INITIAL SAFETY ANALYSIS REPORT

for the New Nuclear Installation Units 3 and 4 at the Temelín Site

Revision 0

Prepared within the meaning of letter A., in Appendix to Act No. 18/1997 Coll., on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act) and on Amendments and Additions to Related Acts, as amended. It is used to meet the condition defined in Section 13(3)c) of the cited act and relating to an application for a licence for siting.

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LIST OF ABBREVIATIONS AND SYMBOLS

Abbreviation	Meaning
AASB	Active and Auxiliary Service Building for Primary Systems
AFIS	Aerodrome Flight Information Service
AFP	Activation and Fission Products
ALARA	As Low As Reasonably Achievable
AMSL	Above Mean Sea Level
ATC	Air Traffic Control
ATWS	Anticipated Transient Without Scram
AUP	Airspace Use Plan
AWS	Automatic Weather Station
BIS	Bid Invitation Specification
BoD	Board of Directors
CAD	Computer Aided Design
CEO	Chief Executive Officer of ČEZ, a. s.
ČEPS	ČEPS, a.s. (Czech Transmission System)
ČHMÚ	Czech Hydrometeorological Institute
CHNR	Cooling Tank with Spraying
CIV	Civil Aviation (General Aviation, Commercial Aviation)
COD_{CR}	Chemical Oxygen Demand measured with dichromate – indicator of water pollution
CoW	Contract of Work
CR	Czech Republic
ČSH	SO 573/01 Pumping Station – civil structure and installed equipment on the left bank of the Vltava River above the dam of the Hněvkovice reservoir
CTR	Control Zone
DEC	Design Extension Conditions
DENA	Specific Density – indicator of geomechanical rock parameters
DGS	Dieselgenerator Station
DNB	Departure From Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DSO	ČEZ Distribuce, a. s.
DV	ČEZ, a. s. Production Division
ECM	Enterprise Content Management – uniform document management system in ČEZ Group
ED_ALT	Young's Modulus – indicator used for geomechanical rock parameters
EG	Engineering Geology

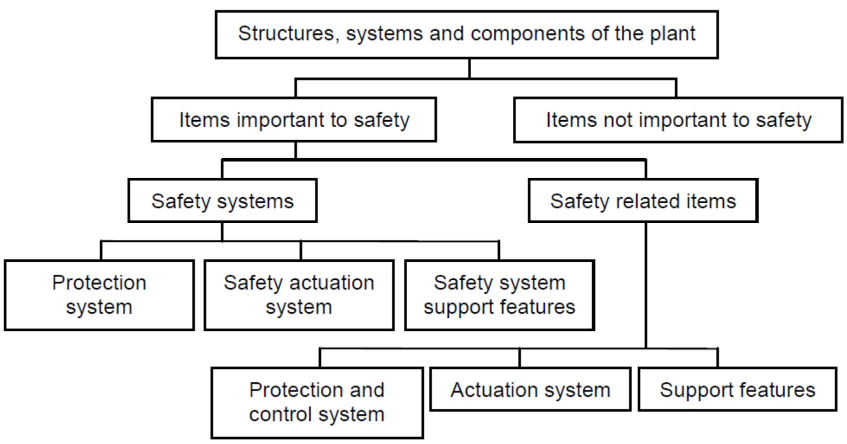
Abbreviation	Meaning
EHV	Extra High Voltage
EIA	Environmental Impact Assessment
EMC	Electromagnetic Compatibility, the ability of equipment or a system to appropriately operate within its electromagnetic environment without creating unacceptable electromagnetic interference for anything in this environment [L. 254]
EMI	Electromagnetic Interference, deterioration in operation of an instrument, equipment or system caused by electromagnetic interference [L. 254]
EMS	Environmental Management System
EP	Emergency Preparedness
EP	Environmental Protection
EPZ	Emergency Planning Zone
ESD	Electrostatic Discharge
ETE1,2	Current Temelín NPP, 2 units with VVER1000 2xVVER1000 reactors
ETE3,4	Future Temelín units 3 & 4
EU	European Union
EUR	European Utility Requirements
FL	Flight Level
FMEA	Failure Mode and Effects Analysis
FP	Fire Protection
ft	Foot – unit of length equal to 0.3048 m
GA	General Avion
GBLR	Generally Binding Legal Regulations
GD_ALT	Shear Modulus (indicator used for geomechanical rock parameters)
GLD	Glider (Sailplane)
HR	Human Resources
HVAC	Heating, Ventilation, and Air Conditioning System
IAEA	International Atomic Energy Agency
IBS	Safety Inspectorate Division of the ČEZ Group (in ČEZ, a. s.)
ICRP	International Commission on Radiological Protection
ICS	Integrated Control System
IFR	Instrumental Flight Rules
IP	Internal Process
IPR	Integrated Pollution Register
ISAR	Initial Safety Analysis Report
ISC	Integrated Subsidiary Company
KP	Commercial-industrial Areas – territory identification in territorial planning documentation

Abbreviation	Meaning
kt	Knot – unit of speed equal to 1.852 km per hour
LAA	Light Aircraft Association
LaC	Limits and Conditions
LPS	Lightning Protection System
LWR	Light Water Reactor
ME	Ministry of the Environment
MEP	Municipality with Extended Powers
MGU	Main Generating Unit
ML, T, LÚ, PM, N	Codes of accident or phase of flight, for their explanation see the legend to Table 47 in this report
MLEEV0, MLEEV, LRSEV, SSEEV	Methods for estimating extreme values of meteorological indicators, see Section 2.4.5.2 of this report
MTOW	Maximum Take-off Weight
NM	Nautical Mile – unit of length equal to 1852 m
NNI	New Nuclear Installation
NPP	Nuclear Power Plant
NPP Temelín	Temelín Nuclear Power Plant – comprised of both power plants: current 2xVVER1000 and future ETE3,4 power plants
NRC	Nuclear Regulatory Commission
NS	Nuclear Safety
OHS	Occupational Health and Safety Management System
PDCA	Plan – Do – Check – Act (Continuous process improvement cycle in quality control systems)
POIS_A	Poisson's Ratio (indicator for geomechanical rock parameters)
POSR	Pre-Operational Safety Report
PP	Physical Protection
PSAR	Preliminary Safety Analysis Report
PVC	Polyvinylchloride
PWR	Pressurised Water Reactor
QAP	Quality Assurance Programme
RAWR	Radioactive Waste Repository
RC	Radiation Control
Rough Levelling	Rough Levelling
RP	Radiation Protection
SDV	Screening Distance Value – distance up to which sources of external human influence are searched for [L. 9]
SFP	Spent Fuel Pool
SI	Safety Instruction

Abbreviation	Meaning
SIGS_K	Unconfined Compressive Strength – indicator of geomechanical rock parameters
SKČ	ČEZ Group
SL-2	Seismic Level 2, see abbreviation "MVZ"
SLZ	Air Recreational Vehicle (ultralight aircraft)
SNF	Spent Nuclear Fuel
SNFR	Spent Nuclear Fuel Repository
SP	Supplier Process (external)
SSE	Maximum Design Earthquake, same meaning as SSE according to US standard NRC 10 CFR 100, Appendix A – maximum design seismic load, during which structures, equipment and systems designed as important for nuclear safety remain functional in terms of the nuclear safety
SSMC	Segment Safety Management Centre
SÚJB	State Office for Nuclear Safety
SÚRO	National Radiation Protection Institute
TLD	Thermoluminescent Dosimeter
TMA	Terminal Control Area – approach and departure controlled airspace
TOC	Total Organic Carbon – indicator of water or air pollution
TRA	Temporary Reserved Airspace
TS	Technical Safety
TSA	Temporary Segregated Airspace
ÚCL	Civil Aviation Authority
ÚJV Řež	Nuclear Research Institute Řež, a. s.
ULL, UL	Ultralight Aircraft
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
VDS	Selected Subsidiary Company of ČEZ, a. s.
VFR	Visual Flight Rules
VHV	Very High Voltage
VJE	Nuclear Power Plant Construction Department
VOJ	Military Aviation Category
VS_ALT	Shear Wave Velocity – indicator of geomechanical rock parameters
VÚV	T. G. Masaryk Water Research Institute
WENRA	Western European Nuclear Regulators Association
WMO	World Meteorological Organization
ZUR JčK	Development Principles of the South Bohemian Region – planning documentation pursuant to Section 36 of Act No. 283/2006 Coll.

GLOSSARY OF TERMS

Term	Meaning
Abnormal operation	An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.
Absorbed dose, D	The fundamental dosimetric quantity D, defined as $D = \frac{d\bar{\varepsilon}}{dm}$ □, where $d\bar{\varepsilon}$ □is the mean energy imparted by ionising radiation to matter with the mass d*m*. The absorbed dose unit in the SI system is joule per kilogram (J kg^{-1}) and its specific name is gray (Gy).
Acceptance criterion	Specified bounds on the value of a functional indicator or condition indicator used to assess the ability of a structure, system or component to perform its design function.
Accident conditions	Deviations from normal operation, which are less frequent and more severe than abnormal operation and include design basis accidents and design extension conditions.
Activity, A	The expectation value of the number of nuclear transformations occurring in a given quantity of material per unit time. The SI unit of activity is per second (s_1) and its special name is becquerel (Bq).
Authorised limit	Mandatory quantitative indicator determined, usually as a result of radiation protection optimisation, for a single radiation activity or a single source of ionising radiation specified by the Office in the relevant licence.
Availability	The capability of systems, structures and components to fulfil the required purpose.
Averted dose	The dose prevented or avoided by the application of a protective measure or set of protective measures, i.e., the difference between the projected dose if the protective measure(s) had not been applied and the expected residual dose.
Becquerel (Bq)	The special name for the SI unit of activity:1 Bq = 1 s^{-1} (~ 2.7 10^{-11} Ci).

Term	Meaning
Certification documentation	Documentation proving and documenting design safety of a facility and qualification of a facility for its intended use in terms of strength, lifetime and seismic resistance. Certification documentation is used as one of the basic documents for licensing, facility qualification, safe operation, modification, conversion, repairs, life extension, etc.
Channel	An arrangement of interconnected components and equipment, required to generate a single protective, control or monitoring signal. A channel terminates at the point where the signal generated by it is combined with signals from other redundant protective, control or monitoring channels.
Classification of systems, structures and components ensuring nuclear safety	All systems, structures and components (hereinafter referred to as equipment) meeting the requirements for ensuring nuclear safety and equipment serving to ensure the technology process of power generation. In terms of nuclear safety, equipment is classified as: equipment which is important to nuclear safety (performing at least one safety function) equipment which is not important to nuclear safety (not performing any safety function). Systems important to nuclear safety are classified by their function and importance to nuclear safety as follows: safety systems safety-related systems  NOTICE: In this context, an „item“ is structure, systém or component

Term	Meaning
Classified equipment	Components or systems of nuclear installations important in terms of nuclear safety, categorised into safety classes depending on their importance to the safety of the nuclear installation operation, on the safety purpose of the system in which they are included, and on the significance of their potential failure.
Collective effective (equivalent) dose	The sum of effective (equivalent) doses to all individuals in a specific group.
Committed effective dose, E(τ)	The sum of the products of the committed equivalent doses to organs or tissues and the relevant tissue weighting factors (w_T), where τ is the integration time in years after intake. The time will be taken to be 50 years for adults after intake and to age 70 years after the intake by children.
Committed equivalent dose, H(τ)	The time integral of the equivalent dose rate in specific tissue or organs to which an individual was exposed after the intake of a radioactive substance, calculated on a model of a reference individual, where τ is the integration time in years.
Common-cause failure	Failure of two or more elements (components) of a system due to a single specific event or cause.
Containment system	A leak-tight building structure and set of elements closing, separating and defining a leak-tight area, passages and penetrations, pressure and temperature reducing and heat removing systems, systems detecting substances in the environment including radioactive material, explosive gas control systems, and ventilation, filtering and other auxiliary systems.
Controlled area	Spaces with controlled access in which special rules exist for ensuring radiation protection or prevention of spreading radioactive contamination.
Controlled state	State of a nuclear installation following an anticipated operational occurrence or accident conditions, n which the fundamental safety functions can be ensured and which can be maintained for a time sufficient to implement provisions to reach a safe state.
Critical population group	A model group comprising these individuals from the population whose exposure relating to a given source of ionising radiation and a given way of exposure is the highest
Defence in depth	A hierarchical deployment of different levels of diverse equipment and procedures to prevent the escalation of anticipated operational occurrences and to maintain the effectiveness of physical barriers placed between a ionising radiation source or radioactive material and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions.

Term	Meaning
Design	The results and the outputs of authorised engineering activity during which a documented design of a nuclear installation is developed and its safety verified or justified, and information necessary for its implementation and on its use is provided.
Design basis accidents	Accidents for which it has been demonstrated, using conservative methodologies in the design, that radiological consequences are kept within the acceptance criteria.
Design basis event	An event considered in the design to be a source of a potential external risk or a combination of external risks considered in the design of a nuclear installation and for which the compliance with the relevant safety requirements has been proven.
Design basis of a nuclear power plant	The range of conditions and events taken into account in the design of a nuclear power plant, and for which it has been proven that the permissible limits will not be exceeded provided that the safety systems will function correctly.
Design criteria	Limiting values of specific physical or process quantities relating to a specific system, building structure or component which, when not exceeded, demonstrably ensure compliance with any of the design requirements in specified design conditions.
Design Extension Conditions (DEC)	Accident conditions that are not considered for design basis accidents, but that are considered in the design process of the plant in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions.
Deterministic effect	Injury in populations of cells, characterised by a threshold dose and an increase in the severity of the reaction as the dose is increased further. Also termed tissue reaction. In some cases, deterministic effects are modifiable by post-irradiation procedures including biological response modifiers.
Deterministic safety analysis	Analysis method that uses, for input parameters of a nuclear installation, specific numerical values, regardless of the probability of their occurrence (taken to have a probability of 1), leading to a single value for the result.
Detriment	Total harm to health experienced by an exposed group and its descendants as a result of the group's exposure to a radiation source. Detriment is a multidimensional concept. Its principal components are the stochastic quantities: probability of attributable fatal cancer induced by radiation, weighted probability of attributable non-fatal cancer induced by radiation, weighted probability of severe hereditary effects, and years of life lost if the harm occurs.

Term	Meaning
Detriment-adjusted risk	The probability of occurrence of a stochastic effect, modified to allow for the different components of the detriment. serving to express the severity of the consequences.
Discharge	Liquid or gaseous substance discharged into the environment that contains radionuclides at a quantity not exceeding the clearance levels or discharged into the environment under conditions specified in the permit to introduce radionuclides into the environment.
Diversification	Design where the safety function is provided by two or more systems which are based on different technologies or physical principles, with a view to eliminating their failure due to a common cause..
Dose commitment, Ec	A calculational tool, defined as the infinite time integral of the per caput dose rate Ė due to a specified event, such as a year of a planned aktivity causing discharges. In the case of indefinite discharges at a constant rate, the maximum annual per caput dose rate Ė in the future for the specified population will be equal to the dose commitment of one year of practice, irrespective of changes in the population size. If the activity causing discharges is continued only over a time period, τ , the maximum future annual per caput dose will be equal to the corresponding truncated dose commitment, defined as $E_c(\tau) = \int_0^{\tau} \dot{E}(t)dt$
Dose constraint	A prospective and source-related restriction on the individual dose from a source, which provides a basic level of protection for the most highly exposed individuals from a source, and serves as an upper bound on the dose in optimisation of protection for that source. For occupational exposures, the dose constraint is a value of individual dose used to limit the range of options considered in the process of optimisation. For public exposure, the dose constraint is an upper bound on the annual doses that members of the public should receive from the planned operation of any controlled source.
Dose constraint	Upper limit of prospective doses to individuals which may result from a defined source, for use at the planning stage in radiation protection whenever optimisation is involved.
Dose equivalent, H	The product of *D* and *Q* at a point in the tissue, where *D* is the absorbed dose and *Q* is the quality factor for the specific radiation at that point, i.e. $H = D*Q$. The unit of the dose equivalent is joule per kg (J kg^{-1}), and its special name is Sievert (Sv).
Dose limit	The value of the effective dose or the equivalent dose to individuals from planned exposure situations that shall not be exceeded.

Term	Meaning
Early release	Situations that could require introduction of measures to protect the public and the environment, however with insufficient time for their adoption.
Effective dose, E	The tissue-weighted sum of the equivalent doses in all specified tissues and organs of the body, given by the expression: $E = \sum_T w_T H_T = \sum_T w_T \sum_R w_R D_{T,R}$ where H_T or w_R $D_{T,R}$ is the equivalent dose in a tissue or organ, T, and wT is the tissue weighting factor. The unit for the effective dose is the same as for absorbed dose, $J.kg^{-1}$, and its special name is sievert (Sv).
Equipment ageing	The process of gradual deterioration of properties of equipment that determine its availability. Ageing is mainly due to the effects of operational conditions, time or changes in the safety requirements. Ageing of equipment of a power plant can reduce the level of safety margins and shorten the total operating time of equipment.
Equivalent dose, H_T	The dose in a tissue or organ T given by: $H_T = \sum_R w_R D_{T,R}$, where $D_{T,R}$ is the mean absorbed dose from radiation R in a tissue or organ T, and w_R is the radiation weighting factor. Since w_R is dimensionless, the unit for the equivalent dose is the same as for absorbed dose, $J.kg^{-1}$, and its special name is sievert (Sv).
Existing exposure situation	A situation that already exists when a decision on control has to be taken, including natural background radiation and residues from past practices.
Fuel assembly	A bundle of fuel elements, fixed in a skeleton frame with which it forms an integral unit. A fuel assembly is transported, stored and loaded into the reactor as a single unit.
Fuel element	A system comprising nuclear fuel, its cladding and other associated components which together form a structural unit. The cladding of a fuel element is the first design barrier against the leak of fission products into the environment.
Fuel element damage	Loss of leak tightness of the fuel element cladding, which is classified as failure of the first design barrier against the leak of radioactive fission products.
Fuel system	Fuels elements, fuel assemblies and reactor core components designed for use in a specific type of nuclear reactor in a specific range of design conditions.

Term	Meaning
Fuel system damage	Situation where the design criteria of any of the components of the fuel system are exceeded, including fuel element damage.
Impermissible exposure	Such exposure, due to which workers or other persons or individuals from the population would or could be exposed to doses exceeding the exposure limits determined in special legal regulations or authorised limits determined in the relevant licence.
Impermissible release of radioactive material or ionising radiation	Release of radioactive material or ionising radiation due to which workers or other persons or individuals from the population would or could be exposed to doses exceeding specified exposure limits or authorised limits set in the relevant licence.
Important protection and control systems	System that monitors the operation of a nuclear installation and which, on detecting abnormal conditions, initiates actions to prevent accident or potential accident conditions. This use of the term "protection" refers to protection of a nuclear installation. The system encompasses all relevant electrical and mechanical equipment and circuitry, from sensors to actuation device input terminals.
Indicative value	An indicator or criterion for assessing the level of radiation protection to be used in the event of unavailability of such detailed information on a specific activity resulting in exposure or on an intervention as would make it possible to assess the optimisation of radiation protection for the specific case.
Ionising radiation doses as low as reasonably achievable	Values optimised in the terms of radiation protection according to applicable legislation (Decree No. 307/2002 Coll. [L. 4]).
Lifetime management	A process combining ageing management with economic planning to optimise the operation, maintenance and service life of systems, structures and components, and to maintain the required level of performance and safety of the unit, and to maximise the return on investment over the planned service life of the power plant.
Limits and conditions	A set of clearly defined conditions, including the permissible range of parameters, requirements for facility availability and setting of protection systems, for which it has been proven that the operation of a nuclear installation in the specified operational mode is safe; they ensure such condition of a nuclear installation, its systems and components as guarantees the validity of the assumptions that have been used in safety design analyses. They also include requirements for personnel activities and for organisational measures to ensure the conditions of safe operation in case they are exceeded.

Term	Meaning
Major leak	Situations that may require the introduction of measures to protect the population and the environment and that may not be limited in space or time.
Non-safety-related systems	Systems not classified as systems important to nuclear safety.
Normal operation	All conditions and planned operations within the operation of a nuclear installation during which the Limits and Conditions for a safe operation of the nuclear installation are complied with; these include, in particular, handling and storage of nuclear fuel, placing the reactor into a critical configuration, steady operation, changes in performance and transients, shutdown, maintenance, repairs and refuelling.
Nuclear installation	Buildings and process units that include a nuclear reactor employing a fission chain reaction; facilities for manufacturing, processing, storage and depositing of nuclear materials, except uranium ore processing plants and uranium concentrate storage; radioactive waste repositories, except repositories containing natural radionuclides only, facilities for storing radioactive waste whose the activity exceeds limits set by an implementing legal regulation.
Nuclear safety	The condition and capability of a nuclear installation and its servicing personnel to prevent uncontrolled development of a fission chain reaction or inadmissible release of radioactive substances and ionising radiation to the environment, and reduce the consequences of accidents.
Operational states	States involving normal and abnormal operation.
Optimisation of radiation protection	Procedures for achieving and maintaining such levels of radiation protection by which the risks to life, human health and environment are as low as reasonably achievable, considering the existing economic and social circumstances.
Personal dose	A summary term for quantities characterising the degree of external and internal exposure of an individual, especially the effective dose, committed effective dose and equivalent doses in each organ or tissue. Personal doses are measured with personal dosimeters.
Planned exposure situation	Everyday situations involving the planned operation of sources including decommissioning, disposal of radioactive waste and rehabilitation of the previously occupied land. Practices in operation are planned exposure situations.

Term	Meaning
Postulated initiating event	An event identified during design as capable of leading to anticipated operational occurrences or accident conditions. The primary causes of postulated initiating events may be credible equipment failures and operator errors (both within and external to the facility) or human induced or natural events.
Potential exposure	Exposure that is not expected to occur with certainty but that may result from an accident at a source or from an event or sequence of events of a probabilistic nature, including equipment failures or operating errors.
Probabilistic safety assessment	A comprehensive, structured approach to the identification of failure scenarios, constituting a conceptual and mathematical tool for deriving numerical estimates of risk.
Projected dose	The dose that would be expected to be incurred if no protective measures were taken.
Radiation accident	An accident that results or may result in intolerable release of radioactive material or ionising radiation to the environment, and which requires measures to protect the population and the environment.
Radiation detriment	A concept used to quantify the harmful effects of radiation exposure in different parts of the body. (It is defined by the ICRP as a function of several factors, including incidence of radiation-induced cancer and heritable diseases, lethality of such conditions, quality of life and years of life lost due to such conditions.)
Radiation emergency	A situation following a radiation accident or such a radiation incident or detection of increased radioactivity or exposure as requires urgent measures to protect individuals.
Radiation incident	An event resulting in intolerable release of radioactive material or ionising radiation or intolerable irradiation of individuals.
Radiation Protection	A system of technical and organisational measures to reduce exposure of humans and protect the environment.
Reactor	A nuclear reactor in nuclear installations as defined in Section 2(h)(1) of Act No. 18/1997 Coll., on peaceful utilisation of nuclear energy and ionising radiation.
Reactor core	A group of fuel assemblies and components in a reactor core.

Term	Meaning
Reasonably achievable level of radiation protection	The reasonably achievable level of radiation protection shall be deemed to be demonstrated and no measure has to be implemented if the costs would exceed the benefit of the measure and if the implementation of the measure requires no special social conditions; the reasonably achievable level of radiation protection shall also be deemed to be demonstrated if the specific radiation activity does not bring about an annual effective dose to any radiation worker exceeding 1 mSv or annual effective dose to any other person exceeding 50 μSv or collective effective dose at workplaces of category IV exceeding 1 Sv, this requirement applying also to foreseeable deviations from normal conditions.
Redundancy	Provision of alternative (identical or different) systems, structures or components so that any one can perform the required function regardless of the state of operation or failure of any other, thus providing protection against single failure.
Reference animals and plants	A reference animal or plant is a hypothetical entity with the assumed basic biological characteristics of a particular type of animal or plant, as described to the generality at the taxonomic level of the family, with defined anatomical, physiological, and life-history properties, that can be used for the purposes of relating exposure to dose and dose to effects for that type of living organism.
Reference level	Index or a criterion whose exceeding or non-meeting leads to adoption of measures in radiation protection; an implementing legal regulation shall lay down details for determination of reference levels and measures adopted as a result of their exceeding.
Release level	Specific activity or total activity level below which radioactive waste, radioactive material and objects or facilities containing or contaminated with radionuclides may be introduced into the environment without the permission of the State Office for Nuclear Safety.
Representative person	An individual receiving a dose that is representative of the most exposed individuals in the population (see Publication 101, ICRP 2006a). This term is equivalent to, and replaces, the term "average member of the critical group" used in previous ICRP Recommendations.

Term	Meaning
Residual dose Basic safety features	A dose that is expected to be delivered even after full application of protective measures (or after a decision not to introduce any protective measures). Basic safety functions of a nuclear installation include: - reactivity control - heat removal from the reactor core and from stored nuclear fuel - isolation of radioactive material, shielding against ionising radiation, control of planned discharges and reduction of emergency discharges of radioactive material
Risk constraint	A prospective and source-related restriction on the individual risk (in the sense of probability of detriment due to a potential exposure) from a source. It provides a basic level of protection for individuals who are exposed to the highest risk from a source and serves as the upper limit of the individual risk in the optimisation of protection from that source. This risk is a function of the probability of an unintended event causing a dose and of the probability of detriment caused by that dose. Risk constraint is a quantity which is similar to the dose constraints, it refers, however, to potential exposures.
Safe state	State of a nuclear installation following an anticipated operational occurrence or accident conditions, in which the fundamental safety functions can be ensured and stably maintained for long time.
Safety culture	The assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance. However, this whole attitude is characterised as an overriding priority.
Safety function	A specific purpose that must be accomplished for nuclear safety.
Safety limit	Limit on an operational parameter within which an authorized facility has been shown to be safe.
Safety related systems	Systems important to nuclear safety other than safety systems. Safety related systems include: protection and control systems high performance systems and structures supporting systems (power supply, cooling, etc.)

Term	Meaning
Safety systems	Systems important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design basis accidents.
Severe accident	Accidents that are not taken into account within the design basis accidents and that lead to severe damage to the fuel system. Some of them are part of the design extension conditions.
Severe damage to the fuel system	Accident conditions during which the basic safety function requiring adequate heat removal is disturbed.
Significant transboundary leak	Leak of radioactive material that may result in exposure of persons or levels of environmental contamination beyond national borders, which exceeds international intervention levels and leads to food and foreign trade controlling measures.
Single failure	A failure which results in the loss of capability of a system or component to perform its intended safety function(s) and any consequential failure(s) which result from it.
Stochastic effects of radiation	Malignant disease and heritable effects for which the probability of an effect occurring, but not its severity, is regarded as a function of dose without threshold.
Supervised area	Spaces subject to continuous supervision for radiation protection purposes.
Systems important to nuclear safety	Systems, structures and components: whose malfunction or failure could lead to unacceptable exposure of personnel or population that prevent design-anticipated events from leading to accident conditions whose functions and properties are designed to manage and mitigate the consequences of design basis accidents.
Validation of computer systems and computational means	The process of testing and evaluating an integrated computer system (hardware plus software) to ensure compliance with functional, performance and interface compatibility requirements.
Verification of computer system and computational means	A process performed at a phase in the system development process to make sure that the correctness and quality requirements imposed within the previous phase have been met.
Virtually excluded conditions	Conditions whose occurrence is demonstrably physically impossible or, with a high level of credibility, is extremely unlikely.

DEFINED STATES OF A NUCLEAR INSTALLATION AS USED IN THIS DOCUMENT

Operational states		Accident conditions	
Normal operation	Abnormal operation	Design basis accidents	Design extension conditions

Note: This categorisation of the states of a nuclear installation into normal operation, abnormal operation, design basis accidents and design extension conditions is based on the IAEA SSR 2/1 document [L. 252].

1 INTRODUCTION AND GENERAL DESCRIPTION OF ECONOMIC AND SAFETY GOALS OF POWER PLANT PROJECTS

1.1 INTRODUCTION TO THE PREPARATION OF THE INITIAL SAFETY ANALYSIS REPORT

1.1.1 PURPOSE

The Initial Safety Analysis Report (ISAR) is a basic document to be submitted to the State Office for Nuclear Safety for assessment as part of an application for the issue of a licence for siting of the proposed nuclear installation under Section 9(1)a) of Act No. 18/1997 Coll., [L. 2] on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (Atomic Act) and on Amendments and Alterations to Some Acts, as amended[1]. The submission of the SAR is required by this act under Section 13(3)c) and shall contain information specified in Appendix A(I).

The primary purpose of the Initial Safety Analysis Report is to provide evidence of suitability of the selected site and the acceptance of siting of the proposed nuclear installation. An affirmative decision for siting of a nuclear installation taken by the State Office for Nuclear Safety (SÚJB) is a prerequisite to the issue of a zoning and planning decision for the construction of a nuclear installation and other licences under the Building Act. In accordance with Appendix A to Act No. 18/1997 Coll. [L. 2], the Initial Safety Analysis Report contains other documents required for the issue of a licence for siting of a nuclear installation. These involve the description and preliminary assessment of design conception from the aspect of requirements laid down in an implementing regulation for nuclear safety, radiation protection, and emergency preparedness; preliminary impact assessment of operation of proposed installation on personnel, the public and the environment; proposal of conception for safe termination of operation; assessment of quality assurance in the process of selection of site and method of quality assurance for preparatory stage of construction and quality assurance principles for linking stages.

1.1.2 HISTORY OF WORK ON THE INITIAL SAFETY ANALYSIS REPORT

As mentioned earlier, the scope and depth of the preparation of the Initial Safety Analysis Report are defined in Appendix A to Act No. 18/1997 Coll. [L. 2]. As this is a rather general definition of documentation preparation, it was decided that the report was to be prepared in several steps, allowing gradual specification of the format and content of the work, using the institute of preliminary consultations with a regulatory body, the State Office for Nuclear Safety. In the first step, the outline of the Initial Safety Analysis Report was prepared, more detailed elaboration of which was based

[1] For acts and regulations cited hereinafter, the text always means a phrase "as amended".

on IAEA GS-G-4.1 [L. 272], as well as the IAEA recommendation determining the format and content of safety analysis reports for nuclear power plants. Work on the preparation of the Initial Safety Analysis Report Study started in January 2007 and the priority goal was to collect and update data to fulfil two basic tasks:

- Summarize all known information about the site, verify its completeness and validity in relation to current requirements for the assessment of the site for siting of a nuclear installation, update in time variable data significant for the assessment and make sure that siting of new units within the historically predetermined area is possible even under current legislative conditions
- Specify and lay down requirements for those characteristics of the new nuclear units that are significant for the assessment of their acceptability at the site with respect to impacts on the public and the environment, not only in operational states, but also in accident conditions

A parallel goal of of the Initial Safety Analysis Report Study was its gradual finalisation in the format, structure and content consistent with the Initial Safety Analysis Report itself to the maximum possible extent in order to provide, when completed, the widest ground for the overall assessment allowing the final specification for the preparation of the Initial Safety Analysis Report.

Subsequently, assessment was carried out both on the level of special departments of an applicant for a licence, and on the level of the institute of preliminary consultations with the State Office for Nuclear Safety, making it possible to formulate the specification for preparing the final version of the Initial Safety Analysis Report. The most important outputs were related to the following areas:

- The content of the introductory chapter of the report was specified and completed. In particular, this included the addition of the parts dealing with the application of safety management principles and the implementation of a safety culture.
- The interrelated settlement of the individual criteria of Decree No. 215/1997 Coll. [L. 1] and criteria according to the standard IAEA NS-R-3 [L. 6] were added to the second chapter relating to the description and evidence of suitability of the selected site with criteria for siting of nuclear facilities and very significant ionising radiation sources
- The content of the third chapter relating to the description and preliminary assessment of basic design criteria for nuclear instalations with respect to nuclear safety radiation protection and emergency preparedness was restructured according to US NRC Regulatory Guide 1.206 [L. 275]. Furthermore, a chapter defining common safety requirements for systems, structures and components (Section 3.3.1 in accordance with the outline) was added. The requirements for systems and functions were mainly derived from the wording of Decree No. 195/1999 Coll. [L. 266], taking into account the safety guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276], IAEA SSR 2/1 [L. 252] and WENRA documentation [L. 27], [L. 270] and, where appropriate, from other documents specified in Chapter 3 hereof.

1.1.3 FORMAT OF THE INITIAL SAFETY ANALYSIS REPORT, GOALS AND DEPTH OF THE PROCESSING OF THE INDIVIDUAL PARTS AND POSSIBLE MUTUAL LINKS

Act No. 18/1997 Coll. [L. 2] requires that the Initial Safety Analysis Report shall be additionally attached to the application for a licence for siting of a nuclear installation, and its appendix defines the content of this documentation as follows:

- Description and evidence of suitability of the selected site from criteria for siting of nuclear facilities and very significant ionising radiation sources as established in a legal implementing regulation
- Description and preliminary assessment of design concept from the aspect of requirements laid down in an implementing regulation for nuclear safety, radiation protection and emergency preparedness
- Preliminary assessment of impact of operation of proposed installation on personnel, the public and the environment
- Draft concept for safe termination of operation
- Assessment of quality assurance in process of selection of site, method of quality assurance for preparatory stage of construction and quality assurance principles for linking stages

When determining the format, content and scope of the Initial Safety Analysis Report, the safety guide of the State Office for Nuclear Safety BN-JB-1.12 [L. 133] was appropriately taken into consideration. The report is introduced with Chapter 1. "Introduction and General Description of Economic and Safety Goals of Power Plant Project", aimed at summarizing and providing basic information relating to the plan. In accordance with the guide, the introduction is prepared and structured according to Section 3.1 of GS-G-4.1 [L. 272], i.e. IAEA recommendation determining the format and content of safety analysis reports for nuclear power plants. In addition to basic identification data relating to the project and its participants, an overview of associated administrative proceedings and their order is provided. For information about the overall scope and expected technical design of the site, the report contains a comparison table for the most import design data of the selected design of PWR that, based on the previous feasibility study, can be considered for implementing a business plan of an applicant for a licence. A bidding process for the selection of contractor of units for the completion of NPP Temelín was organized in accordance with the legislation of the Czech Republic - Act No. 137/2006 Coll. [L. 257] and EU - Directive of the European Commission 2004/17/EC [L. 199]. In addition to the safety requirements, the BIS (Bid Invitation Specification) for the new nuclear installation shall contain other requirements for technical solution, operation and economic criteria.

Chapter 2 describes the site, whilst focusing mainly on data necessary for evaluating the site from the aspect of criteria for siting of nuclear facilities. As there is an extensive set of information collected in connection with the previous construction of ETE 1,2 and with the preparation of the Spent Nuclear Fuel Storage Facility on the premises of NPP Temelín available for Temelín site, references to formerly issued and reviewed reports and surveys that were updated and verified to the necessary extent were used for the preparation of this chapter and for a detailed description of the results of analyses and surveys.

Chapter 3 is focused on the description and preliminary assessment of design concept from the aspect of requirements for nuclear safety, radiation protection, and emergency preparedness. The objective is to define important general requirements of the Investor for technical design of installed units to ensure, while meeting such requirements, acceptability of the project for the site. Furthermore, this chapter contains an analysis of the national and selected international legislative requirements for the project. Comparison allows to make a preliminary assessment of design concept from the aspect of requirements for nuclear safety, radiation protection, and emergency preparedness.

Chapter 4 contains a preliminary assessment of radiological consequences of future operation of the proposed nuclear installation for personnel, the public and the surrounding environment. These radiation effects are also assessed with regard to natural conditions and already existing radiological consequences arising from the operated ETE 1,2.

Chapter 5 presents a draft concept for safe termination of operation and is focused on the principles of safe termination of operation, necessary safety measures and considered methods of decommissioning including time relations.

Chapter 6 assess quality assurance in the process of selection of a site, method of quality assurance for the preparatory stage of construction and quality assurance principles for linking stages.

A list of references, abbreviations and annexes referred to in the individual chapters is added to the text.

1.1.4 ETE3,4 PROJECT PREPARATION

Project identification data

Project name:	New Nuclear Installation "ETE 3,4"
Construction site:	Temelín Nuclear Power Plant,
	South Bohemia Region
	cadastral area of Křtěnov, Březí u Týna nad Vltavou, Temelínec
	Czech Republic

Character of the project

Construction of a new nuclear installation, including associated civil structures and process equipment. In respect of the original concept for Temelín Nuclear Power Plant, the project is to complete the power plant with modern units, including the addition of power lines to the switchyard Kočín.

Initial conceptual documents:

State Energy Concept of the Czech Republic (SEC)

Approved by Resolution No. 211 of the Government of the Czech Republic of 10 March 2004.

The concept is currently being updated; the government took the update of the State Energy Concept of the Czech Republic (hereinafter referred to as the "Concept") under consideration by Resolution No. 803 of the Government of the Czech Republic of 8 November 2012 and approved the introduction of this Concept into the process of environmental impact assessment of concepts (SEA), and at the same time, approved the calculation of a detailed economic analysis and the major elements of energy strategy as formulated in the Concept.

The final wording of the updated Concept together with the result of the process of environmental impact assessment shall be subsequently presented to the Government of the Czech Republic for approval as the update of the existing State Energy Concept.

Territorial Development Policy of the Czech Republic (PÚR)

Approved by Resolution No. 929/2009 of the Government of the Czech Republic of 20 July 2009.

A report concerning the application of PÚR shall be presented to the Government of the Czech Republic at the end of 2012 and the process of updating (or creation of new) PÚR shall be initiated at the beginning of 2013.

Development Principles

Pursuant to Act No. 183/2006 Coll. [L. 256], as from 1 January 2007, the regional office, with delegated powers, shall procure the Development Principles (Subsections 36 – 42) that supersede the Land-use Plan of the Large Territorial Unit of the České Budějovice Residential Regional Agglomeration of 1986 approved by Resolution No. 147/1986 Coll. of the Government of the Czech Socialist Republic of 10 June 1986.

At the beginning of 2007, the recipient of the territorial planning documentation initiated transformation of the approved specification of the Territorial Plan of the Large Territorial Unit of the South Bohemian Region into the form that currently responds to the requirements laid down in the Building Act and its implementing decrees, and includes the conclusions of a conciliation procedure to the concept of this documentation.

The document is called Development Principles of the South Bohemian Region and is embodied in Section 36 of the Building Act. The Development Principles of the South Bohemian Region are prepared for the entire territory of the South Bohemian Region. They deal with the layout of the region and contain plans of national and supralocal importance. They are issued in the form of general measure pursuant to Subsections 171 – 174 of the Administrative Procedure Code that is composed, in addition to the text itself, of statements and justifications.

The Development Principles of the South Bohemian Region respect the approved Territorial Development Policy of the Czech Republic.

The Development Principles of the South Bohemian Region are the basic tool for land use planning activity of the region. The Building Act imposes an obligation on regions to procure and issue such principles by 31 December 2011. The Development Principles of the South Bohemian Region were issued at the 26th session of the South Bohemian Regional Assembly held on 13 September 2011.

The Development Principles of the South Bohemian Region became effective on 7 November 2011 after their proper posting on the official board of all 623 municipalities of the South Bohemian Region as well as on the region's official board.

The Regional Development, Land-use Planning, Building Code and Investment Department of the South Bohemian Regional Authority, as the recipient of the Development Principles, proceeded to the preparation of the report on the application of the Development Principles. Based on this report on the application of the Development Principles of the South Bohemian Region that was discussed and approved by the South Bohemian Regional Assembly on 15 May 2012 under Resolution No. 182/2012/ZK-31, the Development Principles of the South Bohemian Region shall be updated.

Land-use Plan of the Temelín Municipality

The Land-use Plan of the Temelín Municipality approved by the Temelín Town Council by an ordinance of 26 June 1997 has been superseded by a new document. It included design areas for the construction of Temelín NPP with four units VVER 4x1000 MW.

On 26 November 2010, a new Land-use Plan of the Temelín Municipality was issued, which includes the construction of the new nuclear installation in terms of land use, including areas for temporary site facilities.

On 23 February 2012, the Temelín Town Council considered and approved the draft specification of Amendment No. 1 to the Land-use Plan of the Municipality Temelín; Amendment No. 1 is currently being developed.

Land-use Plan of the Dříteň Municipality

Amendment No. 4 to the Land-use Plan of the Dříteň Municipality was approved by the Town Council at its 4th session held on 14 March 2011.

Amendment No. 5 to the Land-use Plan of the Dříteň Municipality, which incorporates areas mainly for the expansion of the switchyard in Kočín and conversions of some electric lines and corridors, was approved by the Dříteň Town Council at its 10th session held on 23 July 2012.

Process of Environmental Impact Assessment (EIA) – major milestones

Elaboration of EIA Notification and its submission to the Ministry of the Environment: 07/2008

Issue of the Conclusion of the fact-finding procedure: 02/2009

Elaboration of EIA Documentation and its submission to the Ministry of the Environment: 05/2010

Preparation of EIA Report: 02/2012

Public hearing of EIA Documentation and EIA Report: 06/2012

Position of the Ministry of the Environment on the Environmental Impact Assessment of the Project:

Assumption 12/2012

Related procedures and expected schedule:

Procedure under the Euratom Treaty

Data concerning the plan under Articles 37 and 41 of the Euratom Treaty

Commencement of the notification:

Assumption Article 41: 09/2013

Article 37: 07/2014

Procedure under the Building Act

Commencement of the elaboration of documentation for Site Decision proceedings (DÚŘ): 10/2013

Filing of the application for the Site Decision: 04/2015

Issue of the Site Decision:

Assumption 10/2015

Commencement of the elaboration of documentation for Construction Permit proceedings for site facilities (DSŘ):

Assumption 11/2015

Filing of the application for the Construction Permit for site facilities:

Assumption 04/2016

Issue of the Construction Permit for site facilities:

Assumption 08/2016

Commencement of the elaboration of documentation for Construction Permit proceedings for main supply (DSŘ):

Assumption 01/2014

Filing of the application for the Construction Permit for Main Generating Unit:

Assumption 01/2017

Issue of the Construction Permit for main supply:

Assumption 06/2017

1.2 INFORMATION ABOUT APPLICANT AND PROCUREMENT FOR POWER PLANT

1.2.1 INFORMATION ABOUT APPLICANT FOR A LICENCE

The joint-stock company ČEZ was established in 1992 by the National Property Fund of the Czech Republic. The major shareholder is the Czech Republic, whose equity share is administered by the Ministry of Finance of the Czech Republic. Main business activity of ČEZ, a. s., is electricity production and sale, and associated support of power systems. ČEZ is also engaged in the production and distribution of heat, as well as the sale of gas and heat. In 2003, a merger of ČEZ, a. s., with distribution companies (Severočeská energetika, Severomoravská energetika, Středočeská energetická, Východočeská energetika and Západočeská energetika) gave rise to ČEZ Group, which became the prominent energy group in the region of Central and Eastern Europe. ČEZ Group ranks among the ten largest energy conglomerates in Europe and is the strongest business entity on the domestic electricity market. In the Czech Republic, ČEZ Group is the largest electricity and heat producer, and a distribution system operator in most regions and the strongest business entity on the wholesale and retail electricity market. Most production capacities are centralized in the controlling company ČEZ, a. s.

Tab. 1 Basic information about the company

Basic information about the company	
Business name:	ČEZ, a. s.
Street:	Duhová 2/1444
City:	Prague 4
Postcode:	140 53
Phone:	211 041 111
Fax:	211 042 001
E-mail:	cez@cez.cz
URL:	www.cez.cz
Sector:	Energy, OKEČ Classification No. 40.10
Company Identification No.:	45274649
VAT number:	CZ45274649
Bank details:	KB Prague 1, account No. 71504011/0100
Core business:	Electricity generation and distribution Heat production and distribution
Auditor:	Ernst & Young Audit s.r.o.
CZECH MADE award:	No
Ratings	A- (Standard & Poor's) of 6 August 2012 A2 (Moody's) of 13 April 2012
Incorporated by entry in the Commercial Register kept by the Municipal Court in Prague, Section B, Entry 1581	

Entrepreneurial Concept of ČEZ Group

The concept reflects the key strategy for the future development of the company in relation to the Energy Concept, which was approved by the Government of the Czech Republic on 10 March 2004 and which outlines a long-term framework for the development of the Czech energy industry.

ČEZ's main objective is to become a key player on the Central European electricity market. This challenge could be mainly reached by acquiring equity shares in power utilities or participating in the projects for new facility construction, especially in the countries of Central and Southeastern Europe. ČEZ is focused both on privatization of state-owned shares in power utilities, and on entry into the companies or projects controlled by private owners. The participation in the projects for new facility construction contributes significantly to the growth of the value of ČEZ.

Corporate Mission and Vision

The mission of the ČEZ joint-stock company is to guarantee a long-term appropriate profit for shareholders through successful business performance especially on both Czech and foreign electricity markets.

The vision of the joint-stock company is to become number one on electricity markets in Central and Southeastern Europe.

ČEZ, a. s., will continue to ensure a stable electricity supply in the Czech Republic, whilst guaranteeing operational safety, and will provide its customers with electricity, including support services, and heating of the highest quality, while a portion of the generated electricity volume will be sold in European markets.

ČEZ, a. s., will maximize the return of capital for its shareholders, provide high-quality and safe services to all its customers, contribute to company development and create an environment for professional growth of its employees.

ČEZ Group Business Line

The primary entrepreneurial activity of the company is the production, purchase, distribution and sale of electricity and the provision of support services.

The main entrepreneurial focus of the ČEZ Group is the production, purchase, distribution and sale of electricity to end customers of all sizes, and the purchase of and trade in gas. This activity generates the predominant volume of costs and revenues. ČEZ's success in this primary entrepreneurial field depends largely on the efficiency of the electricity trade and customer services, in both the wholesale and retail sector. Therefore, the main objective will be to produce an optimum share of electricity generated by its own economically viable sources. Electricity production will be highly reliable, efficient, with controlled safety risks and acceptable environmental impacts. Nuclear safety is the main concern during the production by nuclear installations. The electricity distribution system should provide services indiscriminately, with the required reliability and costs proportional to the regulatory framework. Primary entrepreneurial activities also entail the provision of support services to the transmission network operator, and partly also to the distribution network operators. The objective of the Group is to become a gas trader in order to optimally cover its own needs for gas for newly constructed facilities and subsequently to use the built-up position for further gas trading on the market.

Secondary entrepreneurial activity entails the generation and sale of heat, processing of by-products generated by energy-yielding processes, provision of engineering services and coal mining.

1.2.2 CONTRACTING IN THE SITE DECISION PROCEEDINGS STAGE

Detailed information about contracting in the Site Decision proceedings stage was presented to the State Office for Nuclear Safety in the Quality Assurance Programme for Siting of Nuclear Installation "Units 3 and 4" at Temelín NPP document [P. 10], which was approved by Decision No. SÚJB/OTBIS/16140/2010 of the State Office for Nuclear Safety of: 29 June 2010.

This document outlines in detail processes and activities affecting the nuclear safety or radiation protection. The process of procurement under contract (including nuclear installation); preparation of tender documents, enquiry, bid evaluation, selection of supplier, conclusion of contracts, control and supervision of suppliers during the implementation of the work is identified as a process in direct responsibility of a licensee with direct impact on nuclear safety or radiation protection. An objective is set and powers and responsibilities are described in more detail in the relevant control documentation of ČEZ, a. s., and in the related design documentation of the NPP Construction Department.

The next part of the document specifies the processes and activities carried out by suppliers, again including definition of processes with direct impact on nuclear safety or radiation protection. The document identifies direct suppliers and coverage of activities in processes under contract, provides references to suppliers' quality system documentation, provides references to suppliers' quality system audit reports, certificates, etc.

1.2.3 CONTRACTING IN RELATED STAGES

Detailed information about contracting in the design and construction stage shall be presented to a supervisory body as a part of the presented Quality Assurance Programme for specific stages before their commencement. The establishment of this contracting system shall respect the principles and procedures characterized in Chapter 6 of the Initial Safety Analysis Report.

1.3 BASIC INFORMATION ABOUT THE POWER PLANT

1.3.1 PROJECT OBJECTIVE

The objective of the project is the design, documentation, engineering, know-how, technology transfer, manufacturing and fabrication, supply, construction, erection, testing and commissioning of two complete units of the new nuclear power plants (Units 3 and 4 at the Temelín site), including supply of fuel assemblies to meet the following envelope of parameters:

- Net power more than 1000 MWe;
- Summary probability of significant core degradation lower than 10^{-5}/year;
- Summary probability of large or early release of radioactive materials from the containment or hermetical area lower than 10^{-6}/reactor/year, or in accordance with valid requirements of the State Office for Nuclear Safety;
- Average annual availability higher than 90% for the fuel cycle with a 12-month campaign or more for the first 20 years of operation;
- Minimum designed life of the nuclear power plant unit of 60 years, with no prolonged outage due to component replacement required for at least the first 40 years of its operation;
- Gross design thermal efficiency more than 35%.

1.3.2 GENERAL DESCRIPTION OF THE POWER PLANT

Siting

Completion of Units 3 and 4 of the new nuclear installation is planned at Temelín NPP, about 25 km north of České Budějovice. The power plant will be situated in the area delineated by the Temelín and Březí municipalities and by the villages of Křtěnov and Temelínec. Preliminary delineation of the area for planned completion and orientation of the main structures of the new NPP in relation to the layout of generating units of the existing operated power plant is shown on the Master Plan provided in Annex 1. A possible layout for the completion of two units on the premises of Temelín NPP is shown in Annex 3. The location of the NPP construction site is shown on a land-use map provided in Annex 2, which characterises the vicinity plan.

Based on the assessment of land-use parameters carried out in accordance with legislative and special criteria, the Temelín site was chosen for the location of NPP already in the 1980s. The future construction site is situated in a hilly country, at an altitude of 510 m above sea level. There are no significant elevation points within 10 km from the site. An extensive complex of forests is spread out far away to the northwest. The adjacent area, which is situated on both banks of the Hněvkovice reservoir on the Vltava River, about 5 km east of the site, is also predominantly forested. The power plant will be situated about 45 - 50 km from the state border with Austria and with the Federal Republic of Germany. The closest town is Týn nad Vltavou, located 5 km to the northeast of the power plant.

General Data of the Power Plant

The below brief description of the future design uses a so-called envelope approach for the characteristics of basic parameters of the design, when each design quantity is specified by means of an envelope of parameters, within the limits of which the future design shall range.

The nuclear power plant shall comprise of two units with pressurized water reactors of the selected type, operated in base load mode or in other modes, as agreed upon with the Czech transmission system operator. The gross power of each reactor shall range from 1,198 to 1,750 MWe. A heterogenous pressurized water reactor, with the nominal power of up to 4,500 MWt, shall be used. A flow diagram of each unit shall be two-circuit diagram.

Reactor and Heat Removal System

The primary circuit shall consist of a reactor, a reactor cooling loop, with a reactor coolant system piping, main coolant pumps, compensation system and primary steam generator.

The reactor (specifically the reactor core) shall be cooled and moderated by primary circuit water, pumped through the core by means of main coolant pumps. The reactor core, with an equivalent diameter up to 3.767 m and maximum height of 4.2 m, shall be located in a cylindrical reactor vessel. The reactor pressure vessel and the primary circuit shall be designed for operating pressure 15.5 – 16.2 MPa, at reactor output temperatures in the range from 321 to 330°C. Depending on the type of reactor, the reactor core shall contain 157 - 241 fuel assemblies. The total weight of fuel load shall range from 87 to 144 ton of UO_2. The pressure in the primary circuit shall be maintained by a volume compensation system through coolant heating in the pressurizer or by spraying of the steam blanket in the pressurizer. An adverse pressure increase in the primary circuit shall be suppressed not only through the flexibility of the steam blanket but also by means of a node of discharge and safety valves. Primary coolant activity growth shall be limited by selecting appropriate materials, chemistry and coolant purification in filters. Other related systems shall provide drainage for the main equipment and piping while emptying the primary circuit, controlled collection of primary coolant leaks and their removal into organized leaks, disposal of hydrogen released from primary coolant and purification of gaseous media.

The nuclear fuel shall be contracted under a separated contract together with the main contract for the supply of nuclear units. One of the basic requirements for fuel is its field testing before being loaded into the ETE3,4 reactors.

Secondary Circuit

The secondary circuit comprises steam generation equipment (steam generators), a feed water system, turbine generator, a condensation system, a regeneration system and other auxiliary systems. The steam generators shall generate steam with the range of pressure 5.76 – 7.71 MPa, and at temperatures of 272.78 – 292.5°C, which shall drive the steam turbine and the generator with the range of gross power 1,198 – 1,750 MWe.

The turbine generator is the most important part of the secondary circuit and comprises an extraction condensing steam turbine of a multi-casing design, a collector, electrical generator and an exciting system. The design of the individual systems of the secondary circuit shall meet the safety and reliability requirements, including the necessary back-up condensate and feed water pumps. Turbine bypass capacity shall be sufficient to prevent the reactor trip after turbine shutdown when operating at full power.

Site Layout Concept

The concept of the power plant shall be designed as a single unit, optionally divided into the following main units:

- Nuclear Island (NI)
- Turbine building (PGP, Power Generation Plant)
- Auxiliary systems (BOP, Balance of Plant)
- On-site facilities

Civil structures shall be oriented in the similar way as the current Temelín NPP Units 1 and 2 to allow for the maximum efficient use of the existing infrastructure (power outlet, railway siding, roads, etc.).

The nuclear island of the power plant shall comprise the containment itself and, alternatively, depending on the selected design, active and auxiliary service enclosures and buildings, etc. The nuclear island of the power plant shall be designed to meet the requirements of seismic category 1, without losing the capability to perform the safety functions.

A load-bearing structure of the turbine building shall be composed of a steel scafold. A foundation of the turbine building shall be composed of a reinforced-concrete slab. Both the turbine generator and the structure shall be founded on a reinforced-concrete cast-in-place foundation slab. The orientation of the turbine generator shall direct, to the maximum possible extent, the potential flying objects away from safety-related systems and structures.

Other specific requirements for seismic resistance and resistance to influences of extreme weather arising from site assessment shall be imposed on the selected buildings. The layout among the individual buildings shall be designed so that the individual systems avoid undesirable interaction with safety-related structures and systems.

The required values of seismic acceleration used in designs for buildings of the new nuclear installation shall be in compliance with the requirements specified in Chapter 2.

Safety Systems

The safety systems shall be designed to avoid endangering of nuclear safety in case of failures or accidents, i.e. to shut down the reactor if necessary (stop the fission chain reaction) and maintain it in the safe condition. The safety systems shall also ensure that the residual heat is removed and that no radioactivity is released.

The generation III or III+ safety systems are based on the improvement of the existing system using proven structural elements and available technology innovations to enhance the safety and reliability of the installation. The passive safety systems, which do not require an off-site power supply and are independent of operator activities, are also used increasingly.

The safety concept for the designs is based on the defence in depth principle. The use of the increased number of active and passive technologies results in a decrease in the probability of a severe accident.

The designs include several mechanically and electrically independent circuits of safety systems (preferably concept 4 x 100%), when activity of only one division or safety design with higher percentage of passive elements, which provides a proportional safety profit (AP 1000), shall be sufficient to handle an emergency situation.

A multiple redundancy of safety systems provide for the fulfilment of the safety functions by the last of the four independent circuits when the first one can be put out of operation due to planned maintenance, the second one is not functional due to failure and the third one can be inefficient due to an accident which the system intervenes against. The passive systems shall operate independently of external interventions by operator or technology functions.

Protective Envelope

The protective envelope or containment system shall be designed to eliminate releases to the atmosphere during all operational states as well as in accident conditions and ensure that, if any occur, they take place under controllable conditions in all states of the nuclear power plant taken into consideration in the design. The containment structure and systems shall be further designed to protect the reactor vessel, primary circuit and all associated equipment important in terms of the nuclear and radiation safety, located in the containment, against external events, which occurence cannot be eliminated with sufficient probability. The containment system shall also be designed to provide for the biological shielding function. This means that the structures of the containment system and internals of the primary containment shall reduce direct radiation from all sources contained therein so as to minimise exposure of the public and staff outside the primary containment.

The system shall contain:

- Sealing structures
- Pressure and temperature monitoring and control systems
- Mechanisms for isolation, control and removal of fission products, hydrogen and other materials that could escape into the containment
- Protective (structural/mechanical) elements

A full-pressure primary containment, of a large dry type, with a sufficient free internal capacity in terms of risk of critical hydrogen concentration, rated for all accident conditions considered and a secondary containment shall be used. The required function shall be provided for maximum pressures, underpressures and temperatures as well as other parameters of the accident conditions considered.

Ventilation and Air-Conditioning Systems

New units shall be equipped with the following:

- Ventilation and filtration systems, which in normal and abnormal operation shall:
  - Prevent any dispersion and uncontrolled release of radionuclides in the various areas of the nuclear installation in accordance with the categorisation of these areas
  - Reduce the volume activities of these radionuclides dispersed and leaked into the nuclear installation's accessible areas
  - Maintain the specified climatic conditions
  - Maintain the releases of radionuclides to the atmosphere within the acceptable levels
- Reliable and efficient filters and the possibility of filter efficiency to be tested

For ventilation systems, special attention shall be given to safety aspects such as the inhabitability of control rooms, maintenance of internal climatic conditions, maintenance of pressure zones in the nuclear island and measures to ensure the integrity of fire compartments.

The project shall include special air filtration systems for the main control rooms and, depending on the specific design, for the emergency control rooms and the emergency operation facility protecting the personnel in the case of radioactivity in the air.

Circulation Cooling System

The circulation water system is assumed as a closed-loop system with cooling towers. Since the heat shall be mainly removed to the atmosphere by water evaporation, water losses shall be compensated for from the Hněvkovice reservoir.

Electrical Systems

Description of connection of the new nuclear installation to the power grid of the Czech Republic:

The ETE3,4 units shall be connected to the power grid of the Czech Republic through the 400 kV switchyard in Kočín, to which the ETE1,2 units are currently connected.

The 400 kV switchyard in Kočín is connected to the power grid by five 400 kV lines. Two lines are used for power outlet to the switchyards in Řeporyje and Chodov, another two lines are directed to the 400 kV switchyard in Dasný and one line is connected to the 400 kV border switchyard in Přeštice, which is further connected to the Etzenricht switchyard in Germany. The switchyard in Kočín also contains a 110 kV switchyard, which is fed from the transmission system by 400/110 kV transformation.

The two ETE3,4 units shall be connected to the 400 kV switchyard in Kočín by 400 kV (power outlet) and 110 kV (standby power supply) overhead lines, running in a common corridor. A separate 400 kV line for power outlet and a separate 110 kV line for standby power supply shall be installed for each unit.

Process subsystems for power outlet shall be designed for each unit so that a possibility of transfer and spread of electrical failures between the individual units is sufficiently limited. Upon transformation in generator transformers, the output of every generator or generators of the relevant unit shall be led to the 400 kV transmission system through a grid line. In order to provide high reliability, "two breakers to tap" shall be installed for unit power outlet. A grid breaker for the 400 kV line shall be installed in the switchyard in Kočín.

The standby power supply shall also be designed for each unit and shall provide the required level of functional independency of the auxiliary normal power supply. The 110 kV switchyard in Kočín shall be used as the auxiliary standby power supply for both ETE3,4 units, which also serves as the auxiliary standby power supply for the existing ETE1,2 units.

The construction of ETE3,4 will require reconstruction of the switchyard in Kočín, reinforcement of the external relations for the 400 kV and 110 kV switchyards and, where appropriate, conversions on transmission profiles outside the switchyard in Kočín. The scope of the proposed conversions differs depending on the proposed power alternatives of ETE3,4.

Note: Power outlet diagram shall be specified in the project in relation to the solution of the selected supplier.

The connection of ETE3,4 to the power grid as well as all functions and modes of ETE3,4, including the provision of support services of the new nuclear installation to the power grid, shall be provided in compliance with the Czech Grid Code (ČEPS).

The electrical system within ETE3,4 shall be equipped with the adequate power supplies and grids to provide the below mentioned process and safety functions in terms of power supply.

The tap changing transformers shall be used as the auxiliary normal power supply for each of the generating units. These tap changing transformers could be supplied with power from the generator or through the generator transformer from the 400 kV EHV switchyard in Kočín used for the power outlet of the generating unit.

The standby power supplies shall be used both during the normal and during the abnormal operation as well as in accident conditions in the case of a partial or full loss of the normal power supply. A standby transformer or transformers shall be used as the standby power supply.

In compliance with the basic concept of the nuclear island and the mechanical part, the required scope of the emergency power supply system shall be created.

The emergency power supplies for the category II emergency power supply networks shall comprise of automatically connected diesel generators or other technical means meeting the requirements for the fulfilment of the adequate safety functions.

The emergency power supplies for the category I emergency power supply networks shall comprise of accumulators with uninterrupted power supply systems (rectifiers and inverters).

All power supplies shall be rated to cover the required power levels in normal and abnormal operation as well as in accident conditions.

Note: An ETE3,4 auxiliary diagram shall be elaborated in detail in the preparation phase of the project.

Support services within the meaning of the Grid Code are assumed for all projects. The characteristics and conditions of the connection of the new installation to the transmission and distribution system shall be defined in detail in accordance with the Grid Code valid on the date of conclusion of the Contract of New Nuclear Installation Connection under Act No. 458/2000 Coll. [L. 217] and Decree No. 51/2006 Coll. [L. 282].

I&C Systems Concepts

Each of the ETE3,4 units shall have its own autonomous instrumentation and control systems. The basic systems of I&C architecture shall include:

- Reactor protection system - reactor trip system
- Reactor protection system - systems for activation of safety ensuring technical instruments
- Systems necessary for safe reactor shutdown
- Information systems important for safety
- Other systems important for safety
- Control systems not required for the provision of nuclear safety
- Instrumentation and control diversion systems
- Data transmission systems

The I&C systems shall be designed to contribute to the provision of the necessary characteristics in terms of power generation, manoeuvrability, etc., while maintaining the required safety levels during abnormal operation and in accident conditions.

An advanced technology with an intelligent human-machine interface shall be used to provide high reliability and availability of the generating unit.

In terms of used technologies, advanced, proven technologies shall be used for hardware and software with the possibility of using readily commercially available equipment for the implementation of functions not affecting the nuclear safety. The use of commercially available equipment for safety systems shall be allowed only if such equipment were tested in relevant industrial applications and their compliance with the specific functional, performance and qualification requirements was fully demonstrated.

The project shall extensively apply standardisation and unification for the ease of maintenance, simplification of requirements for spare parts and training.

In terms of system maintainability, equipment architecture shall provide for testability in operation, periodic maintenance, ease of calibration, online diagnostics and self-diagnostic functions. All I&C equipment shall allow for periodic testing without the need for unit shutdown and shall be designed to withstand the environment in places intended for their installation.

1.3.3 CHARACTERISTICS OF MODES AND MANOEUVRABILITY OF UNITS

The applicant for the licence shall submit specific values of parameters listed hereinafter in order to provide the maximum information to the State Office for Nuclear Safety. The indicated values do not comprise an envelope of parameters applicable to the preliminary assessment of the design. The source of data is its requirements for potential suppliers in the BIS, which could be modified during the bidding process.

Reliability

The availability of the power plant is expected to be such that the average annual operating factor is more than 90% for a 20-year reference period at an interval between fuel reloads equal to or longer than 12 months.

Duration and frequency of outages of the installation:

- Planned outages shall not exceed 20 days/year on an average. Refuelling and fuel reloading outage shall be feasible within less than 14 days (breaker-to-breaker),
- The coefficient of (forced) unavailability shall not exceed 1%,
- The main planned outages for periodic total inspections and revisions, overhauls or replacement of large components should not exceed 140 days over the period of 20 years,
- The average frequency of unplanned automatic reactor trip is not expected to be more than once in 7,000 hours of operation at critical parameters.

The design of the power plant shall be able to adapt to refuelling cycles in the range of one to two years.

Life Time

The design of the power plant shall be optimised to reach the planned life time. The life time of all main components of the power plant should be similar as the expected design life. In addition, the power plant shall be designed to allow quick replacement of the selected components and equipment, especially heavy components. Within the preparation of the design, technological procedures for the replacement of such components shall be prepared and the preventive maintenance plan shall be developed based on monitoring of the condition of construction, systems and components where cost-effective.

Manoeuvrability

The design shall provide a high level of operating flexibility of the power plant. The unit is expected to be capable of permanent operation in the range from 50% to 100% of its rated power output. The unit shall be capable of operating in the primary frequency control mode and allow connection to secondary frequency control and active power. The unit shall be capable of participating in voltage control and reactive power balance in the transmission system.

The design of the power plant shall allow operation during planned and unplanned changes of load rate during 90% of the duration of fuel cycle.

The permissible rate of change of unit electric power shall be up to 3% of P_{nom}/min.

The design is expected to ensure the starting time of the unit from cold state (at temperature of reactor coolant below 60°C) to generator synchronising within 48 hours.

It is assumed that the unit shall allow the following number of planned changes of load rate, when each change of load is defined as transition from full power to minimum load and back to full power:

- Twice a day
- Five times a week
- Total 200 times a year

The power plant shall be designed for the number of transients, which must not exceed:

- 3,000 hot starts
- 300 cold starts

The unit shall be capable of disconnecting from the grid by regulating to self consumption during all failures in the grid, which would otherwise lead to power plant shutdown. This includes, but is not limited to, the following failures:

- Voltage deviations
- Frequency deviations
- Loss of synchronisation (stability)
- Irregular current load
- Phase-to-ground faults and short circuits on lines which are not switched off in the period assumed by the design

The unit shall be capable of operating at service load supply for at least 2 hours and shall be designed to withstand at least the following numbers of transients to service load from full power over its design life:

- 90 unplanned transients
- 60 planned transients for periodic tests and commissioning tests

The power plant shall be capable of handling the loss of off-site power (disconnection from the grid) without shutting down the reactor while simultaneously providing self consumption at least during the normal length of the fuel campaign.

1.4 REFERENCE PROJECTS

Reference projects are the possible options for the new nuclear installation to be sited at the Temelín site. The reference projects indicated in Section 1.4.1 are currently considered as possible for the implementation; however, similar projects with PWR reactors from other manufacturers, which shall be in compliance with the envelope of parameters of the plant based on the EIA documentation and with the requirements described in the Initial Safety Analysis Report, are not, in principal, eliminated.

1.4.1 EXAMPLES OF REFERENCE PROJECTS

To meet the objectives of the project, the following options of a nuclear unit using the Generation III or III+ designs, with a pressurized reactor, are taken into consideration for the selection and subsequent implementation:

- European pressurized water reactor EPR
- Pressurized water reactor AP1000
- Pressurized water reactor MIR-1200

1.4.2 MAIN TECHNICAL DATA OF THE REFERENCE PROJECTS

Tab. 2 Main technical data for the new nuclear installation (data for 1 unit)

Design	**EPR**	**AP1000**	**MIR - 1200**
Power Output			
Power, gross [MW_e]	1750	1200	1198
Power, net [MW_e]	1650	1117	1113
Thermal power [MW_t]	4500	3415	3200
Core Circuit			
Main circulation loops	4	2 hot / 4 cold legs	4
Core circuit flow rate [m^3/s]	31.47	19.87	23.9
Operating (nominal) pressure [MPa]	15.5	15.5	16.2
Secondary Circuit			
Steam flow rate at normal conditions [kg/s]	2552	1886	1780
Steam temperature/pressure [°C/MPa]	292.5/7.71	272.78/5.76	286/7
Reactor core			
Core height [m]	4.2	4.267	3.73
Core equivalent diameter [m]	3.767	3.04	3.16
Fuel assemblies	241	157	163
Bundles with absorption elements	89	69	121
Fuel quantity [t UO_2]	144	92.97	87
Fuel cycle length [months]	18	18	18
Reactor pressure vessel			
Cylindrical body inside diameter [mm]	4870	4038.6	4250
Cylindrical body wall thickness [mm]	250	203	200
Total height [mm]	13722	13944	11185
Main coolant pumps			
Number	4	4	4
Nominal flow rate [m^3/h]	28320	17886	21500

Design	EPR	AP1000	MIR - 1200
Pressuriser			
Total volume [m^3]	75	59.5	79
Design pressure [MPa]	17.6	17.1	17.6
Steam generators			
Number	4	2	4
Type	vertical with U-tubes	vertical with U-tubes	horizontal with U-tubes
Maximum outside diameter [mm]	5168	6096	5100
Total height/length [mm]	24621	22460	13820
Inside leak-proof envelope			
Version	prestressed concrete with steel lining	steel	prestressed concrete with steel lining
Volume [m^3]	80000	58333	74169
Outside containment			
Version	reinforced concrete	reinforced concrete	reinforced concrete

The above parameters of the projects are taken from the EIA documentation, from data provided by suppliers during the tender procedure, or from publicly available data from pre-licensing or licensing processes in the country of origin or in an EU country. In the linking stages of licensing process, the individual data shall be specified in accordance with the specific design of units arising from the bidding process for the supplier of the new nuclear installation, while taking into consideration changes resulting from project implementation at the site.

1.5 INITIAL REQUIREMENTS APPLIED TO THE PROJECT IN TERMS OF NUCLEAR SAFETY

1.5.1 PRINCIPAL SAFETY OBJECTIVE

The nuclear power plant shall be designed to ensure fulfilment of the principal safety objective. In compliance with the basic safety principles of IAEA SF-1 [L. 5], the principal safety objective is to protect individuals, society and the environment against adverse effects of ionising radiation. To achieve the safety level that is as high as is reasonably achievable, it is necessary to:

- Prevent uncontrolled exposure of persons and release of radioactive materials into the environment;
- Minimize the probability of those events that could lead to loss of control over the reactor core, over fission chain reaction, the radioactive source or any other source of radiation;
- In the case of such events, handle the situation to minimize their consequences.

The fulfilment of the principal safety objective shall be taken into account in all phases of existence of a nuclear installation, i.e. its planning, siting, designing, manufacturing, fabrication, construction, commissioning, operation up to decommissioning, including transport of radioactive materials and radioactive waste management.

1.5.2 BASIC SAFETY REQUIREMENTS

To meet the principal safety objective, the nuclear power plant shall be constructed in compliance with the legislation of the Czech Republic and with the current internationally accepted safety requirements relevant for nuclear technology. Below listed are requirements considered as binding:

- Acts and implementing decrees of the Czech Republic, including international treaties and conventions by which the Czech Republic is bound
- IAEA safety standards (on the level of basic safety principles and safety requirements IAEA SF-1 [L. 5], IAEA SSR 2/1 [L. 252]) and safety requirements imposed by WENRA [L. 27], [L. 270])

A specific elaboration of basic safety requirements using the listed documents is included in Sections 3.3 to 3.19. In principal, a procedure, which selected the most strict formulation in terms of nuclear safety by comparing in detail the above documents, was used to formulate the individual requirements. More detailed rules for hierarchy of national legislation and international safety requirements, as described hereinafter, shall be used for the specification of further safety requirements and mainly requirements of a lower level.

1.5.3 HIERARCHICAL STRUCTURE OF SAFETY REQUIREMENTS

The system of application of codes and standards is based on the hierarchy of five priority levels, which reflect the applicability of criteria in the nuclear sector as well as other areas. The hierarchy reflects the importance of document levels with an increasing importance from basic to top level. The top level comprises all legal standards binding in the Czech Republic.

The project shall also respect the most advanced requirements imposed by electricity companies, as formulated in the EUR document [L. 264]. These requirements correspond to the current level of science and technology in developed countries and were also prepared in order to simplify and harmonize the permitting process of new nuclear units in EU countries.

If a non-conformance between the provisions contained in documents belonging to different levels should occur, the provision from the document belonging to a higher level shall prevail.

The breakdown of codes and standards applied in the presented proposal of the nuclear installation is depicted in the following figure.

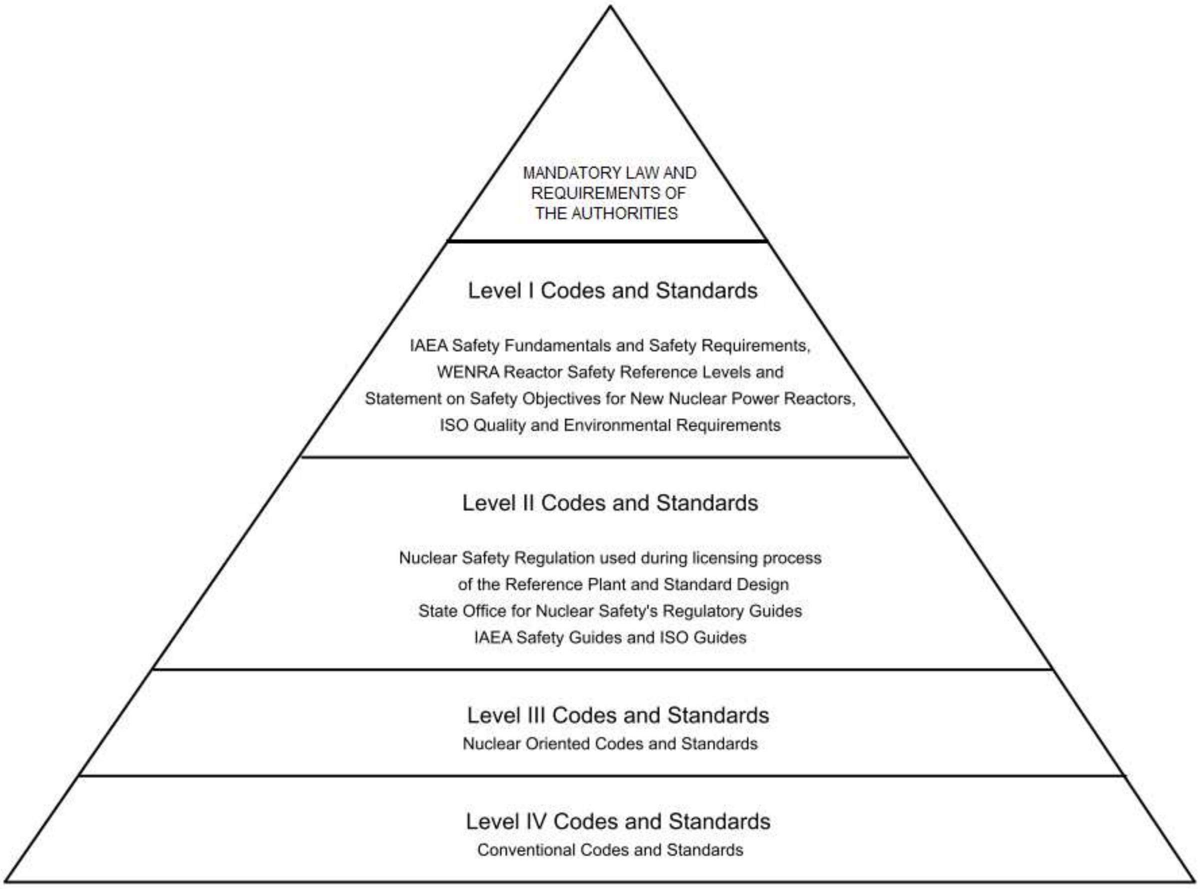


Fig. 1 Hierarchy of legal codes and standards

Mandatory law and requirements of the authorities

This level contains requirements that should be met with no exception. These are the requirements arising from the text of acts, decrees (especially decrees of the State Office for Nuclear Safety) and decrees of the Government of the Czech Republic relating to construction, commissioning, operation and maintenance of the nuclear power plant.

These ČSN standards referred to in technical requirements by provisions of certain acts are also legally binding. The Czech law includes the international treaties and agreements ratified by the Czech Republic, including EU legislation.

Level I – IAEA and WENRA Documents

The Level I includes internal accepted documents, defining basic requirements for nuclear safety. Therefore, any deviation from these documents shall be assessed strictly, case by case.

Design, construction, commissioning and operation of the power plant, including adequate conditions of future safe operation, shall be in compliance with the latest relevant IAEA safety standards on the levels "Safety Fundamentals" and "Safety Requirements" in the following areas:

- General safety principles
- Safety of nuclear installations
- Radiation protection and safety when handling sources of ionising radiation
- Safety during radioactive waste management
- Safety during transport of radioactive materials

This level also includes international standards for quality management in accordance with IAEA and ISO documents.

Special attention shall be given to WENRA documents, which have currently only the nature of recommendation, but these requirements are expected to become the minimum required by law in the future.

Level II – Other Requirements for Nuclear Safety

The Level II of requirements is composed of the following parts - Requirements for nuclear safety in accordance with codes applicable in the country of origin of the project, IAEA Safety Guides, ISO guides and guides issued by the State Office for Nuclear Safety, documents accepted as the international standard and other documents indicated by the client.

The first part of documents relating to nuclear safety covers all fundamentals of safety aspects of the permitting process in the country of origin of the project, supplier or some EU country. These documents relating to nuclear safety shall be applied in compliance with documents containing requirements for systems, structures and components of nuclear installation (Level IV) as a consistent system of codes and standards on which the above required permitting process is based.

If the system of codes and standards in the country, where the permitting process is in progress, fails to contain all necessary special documents or requirements, it is advisable to use the adequate IAEA Safety Standards of Safety Guides series.

Any deviation from the above documents shall be assessed case by case, while the acceptance of the deviation should be closed before concluding the contract.

Level III – Nuclear oriented Codes and Standards

Level III of requirements comprises a complete set of nuclear codes and standards (national, applied in the licensing process in the country of origin of the reactor technology or an EU country and appropriately accompanied with recognized international standards for the nuclear sector, such as ISO, EN, IEC, IEEC, if required) that follow up on Level II. As such, this set of regulations is also the subject of the permitting process and, therefore, required by the client.

As decided by the Supplier, any recognized system of technical codes and standards can be used during designing, supplies of equipment and during construction of the power plant provided that this is not in conflict with the existing Czech legislation. The condition is that this system must be consistent, clearly defined by the supplier in advance and approved by the applicant as well as the nuclear supervisory body. Where possible and practical, the Czech and European standards shall be given priority.

Any deviation from the defined requirements shall be primarily approved in the course of design work. The supplier must demonstrate compliance with documents indicated in Level III. All codes and standards used in the licensing process in the country of origin of the reactor technology or EU country are expected to be applied in the project.

Level IV – Conventional Codes and Standards

Level IV comprises a complete set of general codes and standards suggested in compliance with the previous higher levels, which shall be applied in the design of associated systems of the secondary section and off-site and auxiliary structures of the power plant and in accordance with specific requirements of the client. Any deviation from the defined requirements shall be primarily approved in the course of design work.

It is advisable to apply ČSN and EN standards to the design of the civil part, systems and components of BOP, however not at the expense of the standard design and its suitability. This also means that he supplier can use its national codes and standards, as long as they are not in conflict with the regulations of higher levels; however, the final list of documents of Level IV must be approved by the client.

1.5.4 PRINCIPLES OF APPLICATION OF CODES AND STANDARDS

The final set of requirements applied in the project is expected to be defined accurately in cooperation with the selected supplier in the next stage of safety documents required for the issue of a decision to begin construction. In spite of that, the stipulation of the basic requirements is considered necessary already in this phase of the project, during the selection of a supplier. Therefore, the following rules and requirements shall be applied already in the tender documents, which define to the supplier the client's requirements for supply, in terms of the application of the applicable legal codes and safety standards:

- The supplier shall be obliged to comply with all applicable codes to ensure that the supplied nuclear installation meets the conditions for the obtainment of all necessary permits
- In the country of origin or another EU country, the standard design of the nuclear installation must be at least in such a phase of the permitting process to expect, with high reliability, receipt of a favourable decision issued by the relevant permitting authority
- The bid for the nuclear installation shall contain an accurate identification of the origin of all applied codes and standards, while the applicant shall be entitled to complete them by other documents and approve them before their application in the supply
- All deviations from the set of codes and standards must be consulted with and approved by the applicant; however the supplier shall ensure that no deviations and non-conformities occur or shall be exceptional and justified
- Should the supplier be willing to apply a different codes or standard than those applied in the standard design, the supplier shall document this codes or standard with justification
- The supplier shall be responsible for familiarisation with all relevant codes and standards and for their observance. In this regard, the supplier shall also be responsible for its subcontractors
- In the case of contradiction between the requirements in the approved codes and standards, the principle of applying the most strict requirement shall apply
- The supplier shall notify the client of any doubt concerning the interpretation of codes and standards, while the applicant's statement shall be decisive
- If any new code or standard is developed during construction, the supplier shall notify the applicant thereof together with the specification of their impacts on the project within the defined period of time.

When submitting the bid, the supplier should also consider the relevant documents under preparation, the draft text of which is already known.

1.6 RESERVE

1.7 SAFETY MANAGEMENT PRINCIPLES AND SAFETY CULTURE

1.7.1 GENERAL APPROACH TO SAFETY MANAGEMENT

1.7.1.1 MANAGEMENT IN ČEZ, A. S.

ČEZ, a. s., uses the established, documented, applied, maintained and assessed management system that is continuously improved in order to enhance the safety, quality and protection of the environment. For the basic organisational chart of the ČEZ joint-stock company see Annex 4. The management system of ČEZ, a. s., is oriented on the application and extension of the process approach to management and is composed of basic management areas, management areas and processes. The basic management areas are logically divided into management groups – areas (see Annex 5). A guarantor of the management area is appointed by a guarantor of the basic management area to each management area. The guarantors of the basic management areas are mostly members of the strategic team of ČEZ, a. s., in line management structure.

The management system in ČEZ, a. s., is described in more detail in control and working documents, which comprise an interlinked system. The documentation system contains a description of planned and systematic activities required for ensuring an adequate trust that the requirements for the required level of management have been met.

The established quality system of ČEZ, a. s., (licensee) ensures that requirements in accordance with Decree No. 132/2008 Coll. [L. 258] are assigned to each process, affecting nuclear safety and radiation protection, which is part of the basic management area or management area, and fulfilled. This system is described in binding control and working documents of the quality system (guides, procedures, methods and other working documents).

Requirements for the management system are specified in standard type document [P. 3]. The integrated control system is established in ČEZ, a. s., in order to integrate the principles and requirements for quality, the environment and safety into the management system. The main objective of the integrated control system is to ensure the fulfilment of the company's strategic objectives and secure efficient functioning of the integrated control system as a tool for continuous improvement of quality and level of meeting the requirements for safety and the environment.

The integrated control system in ČEZ, a. s., is established in compliance with the requirements laid down by Decree No. 132/2008 Coll. [L. 258] for the following licensed activities in accordance with the Atomic Act:

- Operation of a nuclear installation or a workplace of category III or IV
- Restart of a nuclear reactor to criticality following a fuel reload
- Reconstruction or other changes affecting nuclear safety, radiation protection, physical protection and emergency preparedness of a nuclear installation or a workplace of category III or IV
- Radioactive waste management in the scope and manners defined by the implementing regulation
- Professional training of selected personnel (Section 18(5))

The concerned basic management areas of ČEZ, a. s., where the above licensed activities are carried out, are as follows:

B – Safety

N – Purchase and Sales

P – Human Resources

R – Management and Administration

V – Production

I – Engineering

The established integrated control system of ČEZ, a.s., respecting the requirements laid down by Decree No. 132/2008 Coll. [L. 258], is appropriately applied in the individual stages of the implementation of the new nuclear installation project (siting, construction, commissioning – inactive, physical, power start-up and trial operation) and documented in the relevant quality assurance programme for the specific stage.

These activities are included in the basic management area of ČEZ, a. s.:

T – Investments

Safety in Management System of ČEZ, a. s.

In ČEZ, a. s., safety plays a key role in the management system. The safety requirements are always met first compared to the other requirements, taking into account the adequacy of spent costs and the risk level. The Safety and Environmental Protection Policy is the top document applying this principle. The safety elements are applied in the part of strategic priority in an order issued by CEO on an annual basis (ČEZ CEO's Strategic Order).

The rule [P. 11] is the basic document describing the safety management system in ČEZ, a. s.

Furthermore, the principles identifying general objectives and basic safety principles that are applied in ČEZ, a. s., are defined in the guide [P. 35], which is binding upon all employees of ČEZ, a. s., and after the application of a contract, upon supplier personnel during activities carried out on installations belonging to ČEZ, a. s.

Graded Approach

A graded approach is established in the company The graded approach for some items is established based on the requirement stipulated by legislation and for others it is established based on design approaches and/or is contained in the relevant standards used.

The graded approach takes into account the following:

- Adequate resources, taking into account the importance and complexity of an item
- Risks and the scope of potential impact on safety, health and the environment
- Process complexity, activities, their inputs and outputs, and their importance from the viewpoint of nuclear safety

- Classification and records of sources of ionising radiation, method of their handling and categories of workplaces, where radiation-related activities are carried out
- Method of nuclear material and radioactive waste management defined by legal regulation
- Categorisation of selected equipment into safety classes (into a list of selected equipment including a list of selected equipment specially designed)

The graded approach in ČEZ, a. s., is not in conflict with the requirements for a "conservative approach". In the cases when any non-conformity occurred (was detected), having impact on the nuclear safety, radiation protection, physical protection and emergency preparedness, always take such action to minimise risks, even at the expense of potential economic losses.

In the company, the graded approach is established across nuclear activities, especially in basic management areas "V" (classification of components and systems, approaches in operating documents, modes, Limits and Conditions, etc.), "B" (different levels of control, supervision, review, categorisation, for example, of waste, application of the ALARA principle, Probabilistic Safety Assessment, further measures to quantify risk and subsequent management of activities make the risk as low as reasonably achievable, etc.), "R" (strategic decisions, system settings, different levels of review, control and assessment, organisation and assessment of organisational changes, higher intensity of communication about safety-related items), "I" (design basis, levels of review and independent assessments, increased engineering support for safety-related items, use of advanced technologies, etc.), "P" (increased requirements for personnel qualification, increased requirements for personnel assessment) and "N" (increased requirements for suppliers, their assessment and verification, increased requirements for supplier controls, transfer of increased requirements for purchased items into contracts and other purchasing documents,etc.).

Safety and Environmental Protection Policy and Management Quality Policy

The strategic management undertook to establish, maintain, assess and continuously improve the safety, environmental protection and quality management system. A Safety and Environmental Protection Policy was issued in order to create general conditions in the area of safety and to fulfil the missions and business plans of ČEZ Group. A Management Quality Policy was issued in order to create general conditions for management and to fulfil the missions and business plans of ČEZ Group.

The principles of safety, safety culture, customer satisfaction, meeting of needs and expectations of the public are applied in the policies, including observance of the applicable legislation and international commitments of the Czech Republic.

In these documents, the Board of Directors accepts a commitment (in compliance with the applicable legislation and international commitments) to:

- Ensure safety of production resources of the company
- Provide protection to individuals, society and the public
- Provide environmental protection
- Assure quality

In addition, the Board of Directors undertakes to develop the adequate conditions and sufficient human and financial resources, efficient management structures and inspection mechanisms to meet the above mentioned commitment.

The principal objective in the area of human resource management and key areas in the area of personnel policy, including partial objectives and applied principles, is defined in the document [P. 36].

Corporate Culture

The company's corporate culture is managed within the basic management area P - Human Resources.

ČEZ's corporate culture is based on the company's strategy and represents a uniform platform for common sharing of basic corporate values used for the derivation of all principles, standards and formulas for the expected behaviour. Seven corporate principles that interpret the nature, philosophy and approach to all activities are an important tool that supports the achievement of set high-level objectives. They identify the method of interpretation of each decision, order or instruction. They also identify behaviour towards personnel, customers as well as business partners.

The main principles of the corporate culture (7 principles) are communicated within the company, e.g. via ČEZ Group Intranet, as follows:

1. The highest priority is to create values, while always maintaining safety (we create values safely).
2. We are all personally responsible for achieving results (we are responsible for results).
3. The actions of each of us must lead to the benefit of ČEZ Group (we are one team).
4. We constantly work on our professional as well as personal development (we work on ourselves).
5. We create an international company (we expand across the borders).
6. We are open to changes and accept better solutions (we look for new solutions).
7. We are fair and loyal to our principles and company (we act fairly).

1.7.1.2 CHARACTERISTICS OF THE APPROACH OF ČEZ, A. S., TO PROCESSES UNDER CONTRACT

Supplier relations are set in the processes of basic management area N and in management areas V04 (governed by guide [P. 39]), P04 (governed by document [P. 44]) and R04 (governed by guide [P. 4]). The mission of the management area V04 – Optimisation of Requirements for the Supplier System is to manage the supplier system to ensure stability, efficiency, quality and safety when performing activities and defining strategies in the area of the supplier system. Supplier relations are strictly regulated by contracts of work and control documents issued by ČEZ, a. s.

Suppliers providing products for ČEZ, a. s., (governed by specific contracts of work) are registered in the relevant system of monitoring business partners. Introduction of new business partners, modifications, cancellation of business partners, etc., shall be governed by procedures linked to the document [P. 15] and guide [P. 14].

The graded management approach applied to suppliers and to the purchased product is defined by the principle of ensuring the safety and high quality of procedure items, while the highest priority in the area of nuclear energy, specifically the nuclear safety, and the observance of the required quality category and the required quality level of items in all areas are always respected.

Contracted Processes

In the cases when the processes or any part thereof (activities) are implemented under contract, their implementation is ensured by a guarantor of the relevant process through a contractual relationship. For safety-related items, the performance of the contracted process shall be supervised by the licensee.

Supervision of the processes or any part thereof (activities) under contract performed by the licensee shall be documented in the form of customer audits and assessment of supplier activities:

- Customer audits of suppliers

  The objective of customer audits of suppliers of ČEZ, a. s., is to systematically verify the competency and qualification of the existing as well as the potential suppliers in compliance with customer specifications, legal regulations, harmonized technical standards and codes. The graded approach in verifying suppliers/subcontractors in accordance with the safety relevance of the subject of the provided supplies and/or services is applied to achieve this objective. Customer audits of suppliers are described in the document [P. 24].

- Evaluation of suppliers

  The objective of the evaluation of suppliers with regard to the safety relevance of the supplied items is to continuously monitor the performance of processes and activities under contract in accordance with the predefined criteria.

  The special departments concerned (guarantors of processes) in the scope of their operation shall be responsible for definition of evaluation criteria and performance of a systematic supervision in compliance with the specified procedure.

  The evaluation of suppliers is described in the document [P. 17].

Supplier Selection

Suppliers of externally supplied items are selected in accordance with the necessary references and information concerning the overall competency of the business partner, especially in the area of quality, reliability and safety of items, management system and the overall financial, assets and commercial situation.The highest priority is to ensure nuclear safety and high quality in the area of nuclear energy, and to observe the required quality category and the required quality level of items. The selected suppliers are included in the Business Partner Database. This procedure is described in the guide [P. 14].

Contractual Provisions Relating to Non-Conformities

The contracts also include the requirements for information about anomalies, i.e. supplier's (seller's) obligation to notify the consumer about any existing or potential non-compliance of the achieved state with the defined quality criteria, including, but not limited to, problems related to deadlines, quantities, etc., including definition of the form of consideration and approval of proposed measures to eliminate such anomalies. In addition, they contain requirements for warranty periods for the subject of the performance including characteristics of the obligation met, the method for lodging claims arising out of the liability for defects in performance, etc. This procedure is described in the guide [P. 14].

Process of Verification of Supplier Competency

The process of verification of supplier competency is based on the following:

- Correct identification of an item with impact on nuclear safety, technical safety and radiation protection
- Keeping of an up-to-date list of approved suppliers (conditioned by valid customer audit and evaluation in 1 – 2 class (complies – complies with minor deficiencies))
- System of evaluation of suppliers:
    - Commercial evaluation
    - Technical evaluation
    - Safety evaluation
- Customer audit process
- Consistent supervision of radiation protection of supplier personnel

Principles of Assessing Supplier Competency

The following principles are applied in assessing supplier competency:

- Use of the results of supplier verification – consumer audit
- Use of the results of evaluation and available data obtained from evaluation
- Provision of feedback to suppliers as a basis for improvement
- Use of the graded approach in assessing supplier competency, especially impact of supply on nuclear safety, technical safety and radiation protection
- Recording in verifying / assessing supplier competency
- Fulfilment of the principle of mutual awareness – information sharing
- Verification of the entire supplier chain, primarily direct supplier / manufacturer of an item, contract partners, only in terms of risks occurring due to added value from intermediation
- Activities relating to assessment and verification of competency are described in the methodology [P. 18]

1.7.2 SPECIFIC ASPECTS OF MANAGEMENT OF SAFETY PROCESSES

1.7.2.1 SAFETY MANAGEMENT IN ČEZ GROUP

There are three levels of safety management defined in ČEZ Group that are compatible with the levels in line management:

- **Group** – feedback for strategic management and coordination of supra-segment / intersegment activities related to safety management
- **Segment** – safety management in segments of ČEZ Group

- **Performance** – performance management / performance of activities to assure safety on the level of organisational divisions in ČEZ, a. s., integrated subsidiary company and selected subsidiary company

Divisions in ČEZ, a. s., integrated subsidiary company and selected subsidiary company are assigned to one of two Segment Safety Management Centres.

Grouping of divisions in ČEZ, a. s., integrated subsidiary company and selected subsidiary company into a segment is based on the principle that type-similar risks are handled within the segment – associated with the generation of heat and electricity or distribution of electricity, and one of the divisions or companies within the segment is:

- Major carrier of risks
- Carrier of know-how in safety management
- Dominant customer of such grouped divisions or companies

and this company or division carries out the function of the Segment Safety Management Centre.

The Segment Safety Management Centres in ČEZ Group are created in:

- Production Division
- Distribution and Foreign Affairs Division

Tools of Safety Management System in ČEZ Group

The safety management system complies with the requirements defined by rules [P. 1] and other top control documents of ČEZ Group or ČEZ in the basic management system R - Management and Administration and is aimed at meeting the commitments and expectations defined in the Safety Policy.

The safety management includes:

- Definition of binding requirements imposed by legislation, which should be/have to be met, i.e. all requirements imposed by legislation (general binding legal regulations) and selected recommendations of "authorities" in the area of safety (IAEA, WANO, EMS, Safe Plant, etc.)
- Transfer of these requirements into specific control, regulatory and working documents of the individual entities
- Control, supervision and monitoring of the fulfilment of such requirements
- Adoption of precautionary measures when an opportunity to improve the safety level has been found
- Adoption of measures in the event of failure to meet such requirements

The relevant line manager shall be primarily responsible for the safety within the meaning of fulfilment of the defined safety requirements of the entity (company, division, organisational unit,…).

The above mentioned provision shall not relieve any worker from his/her responsibility for meeting the safety requirements.

Safety and Environmental Protection Policy

The Safety Policy presents the current commitments and expectations of company management for safety management system. It is linked to a strategic temple of ČEZ Group and it is a prerequisite for the function of all its elements. It elaborates the safety principles from all areas of safety and environmental protection with special emphasis placed on the area of nuclear safety, safety culture principles in compliance with IAEA and WANO recommendations and corporate principles, mainly the "We create values safely" principle based on the requirement for safety risk analysis and assessment. The Safety Policy is approved by the Board of Directors of ČEZ, which entrusts CEO with its issue. All employees of ČEZ, the integrated subsidiary company and the selected subsidiary company are responsible for the fulfilment of commitments and expectations of the Safety Policy. The adoption of binding force in the integrated subsidiary company and selected subsidiary company is ensured by executors of material management of ownership interests. The Safety Policy is a framework for control documents, internal standard for sharing the best practice and management system development.

Safety Policy Structure

The Safety Policy is structured as follows:

- Company commitments, i.e. commitments of the top management of the company towards the parties involved
- Expectations of company management, i.e. expectations of the top management of the company towards company personnel to fulfil the commitments of the company (expectations from company personnel)

Work with Safety Policy

- Announcement of safety-related topics for the current year - announced by the CEO on the basis of a proposal from the Safety Inspectorate Division in the form of annual tasks in relation to key performance indicators by priorities of the individual divisions of ČEZ, a. s., the integrated subsidiary company and the selected subsidiary company
- The divisions of ČEZ, a. s., the integrated subsidiary company and the selected subsidiary company shall elaborate the safety-related topic, in the form of orders issued by their managers, for measurable objectives for the relevant year completed by items from internal feedback
- The Safety Inspectorate Division in cooperation with the Segment Safety Management Centre shall perform topic-oriented inspections in the extent reasonable to the activities carried out, focused on safety-related topic of the year and it evaluates any weaknesses
- The Safety Inspectorate Division shall prepare annual reports on the state of safety in ČEZ Group
- The Safety Inspectorate Division shall suggest safety-related topics for the next year based on the evaluation of the annual report on the state of safety in ČEZ Group
- The Segment Safety Management Centre shall be responsible for the implementation of the Safety Policy on the performance level

Independent Feedback

An independent feedback is provided on the group level for strategic management of the Safety Inspectorate Division and on segment levels of safety management for the management of the Segment Safety Management Centre by the Safety Department in the Production Division and by Integrated Quality Management in ČEZ Distribuce, a. s.

1.7.2.2 SAFETY CULTURE

The safety culture forms an integral part of the corporate culture. Safety culture principles are incorporated in the Safety Policy and are enforced together with other commitments and expectations by employees in managerial positions on all levels. The safety culture is an indicator of the level of implementation and adoption of safety standards or safety organisation in the company, when weaknesses are considered as starting points and opportunities and room for improvement. The safety culture is described by the following principles:

WANO Principles

- Everyone is personally responsible for safety
- Leaders demonstrate their commitments to safety
- An atmosphere of mutual trust is established
- Decision-making reflects the "safety first" principle
- A questioning attitude is applied in the decision-making process
- Nuclear technology is recognized as special and unique
- We learn from mistakes (learning organisation)
- Nuclear safety undergoes constant examination

IAEA Principles

- Safety is a recognised and clearly identifiable value
- The route to safety is obvious
- Responsibility for safety is clearly defined
- Safety integration – is part of all activities
- Learning enhances safety

The management system supports a strong safety culture by:

- Providing general comprehensibility of key aspects of the safety culture
- Providing any means through which it supports individuals and teams in safe and successful performance of their tasks and taking into account mutual interaction among individuals, technologies and processes
- Strengthening the attitude to obtainment of knowledge and questioning on all levels
- Strengthening the questioning attitude and effort to achieve the characteristics of a learning company on all levels

- Providing any means through which it sustainably seeks to develop and improve the safety culture

Examination of Safety Culture

The level of safety culture is periodically reviewed and measures to sustainable improvement are adopted. The period, extent and method of examination are defined by the Safety Inspectorate Division in cooperation with the Segment Safety Management Centre in accordance with the methodology [P. 45].

Application of Safety Culture

The safety culture is incorporated in the management system of ČEZ, a. s., as one of the tools to achieve a sustainable high level of safety.

Personnel as well as suppliers are informed about the principles and key aspects of the safety culture within the basic preparation as well as other educational activities (training, fellowships, courses, etc.), see the document [P. 44] and related documents. Periodic training of all employees in managerial positions in the area of safety and quality is specified by qualification requirements, when a significant part of this training addresses the principles and requirements relating to the safety culture.

ČEZ, a. s., focuses on the prevention of human failure during safe and successful performance of tasks, taking into account the mutual interaction among individuals, technologies and processes. Therefore, the programmes for improving human performance quality are implemented on nuclear power plants.

The level of safety culture in ČEZ, a. s., is assessed by means of repeated examinations, and for nuclear activities, the safety culture is additionally assessed externally, e.g. within OSART or WANO missions. The results of the assessment of the safety culture form the required feedback for all management levels.

Based on the last examination of the level of safety culture in 2011 and its evaluation, the Board of Directors approved an action plan for enhancing the corporate culture and the safety culture. This document defines specific tasks, dates of performance and responsibilities to all materially relevant employees in managerial positions for the areas with potential improvement, especially the area of awareness and experience sharing. The expectation of the top management of ČEZ, a. s., to utilise knowledge and obtained experience for sustainable improvement (characteristics of learning company) on all levels is emphasized in the Safety and Environmental Protection Policy of ČEZ Group.

With regard to the relation of the safety culture to the corporate culture of ČEZ, a. s., the enhancement of safety culture is also supported by activities focused on improvement of corporate culture. Nuclear power plants support the improvement of safety culture by special programmes, such as Safely 15 or 16 Tera. All resources allocated to such programmes are monitored on the level of the top management of ČEZ, a. s.

Committee on Safety of Installations of ČEZ, a. s.

This committee is established by order issued by CEO of ČEZ, a. s., as its advisory board, see document [P. 46]. Its activity is defined by the statute of the committee.

1.7.2.3 IMPLEMENTATION OF SAFETY MANAGEMENT AND ELEMENTS PERTAINING TO SAFETY CULTURE MANAGEMENT IN THE COURSE OF THE NEW NUCLEAR INSTALLATION CONSTRUCTION PROJECT

The principles of safety management and the safety culture elements implementation are applied in the construction project of the new nuclear installation in accordance with the aforementioned corporate approaches within the framework of the Segment Safety Management Centre within the Production Division. Based on the general binding legal regulations and selected international recommendations, the Safety Management Segment Centre defines safety requirements. A safety manager is appointed for the Project in the segment level - Quality and Safety Manager, who cooperates on the creation of safety requirements and oversees their fulfilment.

During implementation, a graded approach enabling to take into account the complexity of the process, activities, inputs and outputs thereof and importance thereof from the point of view of nuclear safety is used. In accordance with the newly acquired international experience, the application of the safety management enabling the implementation of the safety culture principles within the graded extent, depth and scope, starting with the preparation of the units construction, is assumed.

The aforementioned means that:

- Binding safety requirements, which must be adhered to in the period of the units construction, shall be defined
- Such requirements shall be incorporated in the specific control and working documents of the individual entities participating in the construction
- Inspection, supervision and monitoring of adherence to such requirements, including the preparation of proofs (such as protocols) substantiating the fulfilment of the requirements, shall be performed
- Preventive corrective measures shall be taken in the case deviations are discovered with a view to improving the level of safety
- The graded approach shall be applied

The common safety policy shall be defined and accepted on the part of all key entities providing for the construction (as a priority, on the part of the contractor of the nuclear island and the applicant for licence),which shall contain the following:

- Common commitments concerning the project management
- Common expectations of the project management, i.e. expectations of the top project management towards the employees of all entities participating in the construction of the new nuclear units in respect of the fulfilment of the safety-related commitments concerning the project (expectations of the employees of the companies participating in the construction project)

For the purpose of implementation of the required degree of the safety culture in the course of the construction, a methodical programme including the entire spectrum of activities, starting with induction training prepared within the necessary scope for all employees who start working on the project, up to the coaching of the entities operating in the subcontractor chains, shall be prepared. Measurable indicators characterizing the degree of the achieved safety culture concerning specific activities of the construction shall be defined within the necessary scope. A continual system allowing methodical evaluation thereof shall be defined, as feedback for the top project management, to enable continual improvement of the implemented system.

Reporting (within the necessary scope) shall be implemented.

Emphasis shall be put on the fact that all employees participating in the project are aware of the safety relevance of the activities performed by them.

Emphasis shall be put on the fact that there is an easy way enabling the provision of information on any deviation being safety-relevant from the lowest level of the work performance to the respective management level competent to provide for remedy.

A robust system of non-conformities monitoring enabling keeping on file, categorization, monitoring of non-conformities throughout the entire cycle from the occurrence thereof up to its provable removal shall be implemented.

It shall be assured that the programme of implementation of the safety culture principles shall be accepted by the managements of all entities participating in the safety-relevant activities during construction and that the programme shall be applied by them on a continual and active basis by means of examples and direct work with the managed employees performing the works.

1.7.3 SAFETY CULTURE

1.7.3.1 SAFETY AND QUALITY POLICY AND SAFETY OBJECTIVES OF ČEZ, A. S.

The safety and environmental protection policy and the management quality policy were declared by the Board of Directors of ČEZ, a. s., and represent the top safety-oriented management documents of the company. The problems related to safety, quality and environmental protection comprise an interconnected complex. Therefore, these areas within ČEZ are linked to each other also in terms of integrated management.

Nuclear safety assurance is built on the functional management system, qualified personnel and top-quality technologies. These three areas are then system-controlled and continuously improved to meet not only legal requirements but something more, leading to the implementation of the best international practice and the current level of science and technology. For a detailed specification see the document [P. 50].

In the management quality policy, the Board of Directors of ČEZ, a. s., acknowledges full understanding that the quality as the level of the fulfilment of needs and expectations is a decisive safety and competitiveness factor. The quality forms a part of the social responsibility and is a task of every individual in ČEZ Group.

The quality may only be achieved by continuously creating the environment for it. Therefore, strategic objectives are set as a framework for long-term planning and the necessary financial and human resources are provided. Opportunities for improvement are continuously sought and applied, and knowledge, skills and experience are developed. The Board of Directors declares by example to lead employees towards the fulfilment of objectives and towards conduct in accordance with the principles of corporate culture.

ČEZ management sees the quality as observance of the following principles:

- A partner and customer approach is applied.

- Planning takes place in accordance with strategic objectives.
- The best practice is standardised and described.
- Jobs are performed without error and on the first attempt.
- A continuous inspection activity is carried out and discrepancies are immediately addressed.
- Decisions are made based on knowledge of the matter and verified facts.
- A continuous improvement is carried out and changes are executed in a flexible and safe manner.

Safety Objectives in the Area of Operation of ČEZ Group

The Safety Policy of ČEZ Group is implemented by fulfilling the following stipulated safety objectives and observing a number of safety principles that are applied by a graded approach depending on the severity of risk and the magnitude of possible consequences.

Nuclear Safety

Achieve the adequate operating conditions, prevent incidents/accidents and mitigate their consequences with a view to protecting workers and the population against the risks of ionising radiation from a nuclear installation.

Radiation Protection

Set the system of technical and organisational measures to reduce the exposure of natural persons and protect the environment.

Technical Safety

Provide the capacity not to endanger human health and property under specified conditions throughout the lifetime of the selected equipment and ensure consistent compliance with technical requirements contained in an implementing legal regulation or another binding specification for the selected equipment.

Fire Protection

Minimize the probability of fire occurrence and propagation and reduce the possible consequences to the lowest acceptable level.

Physical Protection of Nuclear Materials and Nuclear Installations

Set the system of technical and organisational measures preventing unauthorised operations using nuclear facilities, nuclear materials and selected items.

Occupational Health and Safety

Minimize adverse impacts of operating and manufacturing processes on health of employees and other persons.

Protection of the Environment

Minimize adverse environmental impacts of operating and manufacturing processes.

1.7.3.2 APPROACH TO MEASUREMENT, ASSESSMENT AND IMPROVEMENT IN ČEZ, A. S.

The individual levels of monitoring and assessment in ČEZ, a. s., are described in the document [P. 40]. The system of measurement, assessment and improvement applies the following levels:

- Own assessment
- Independent assessment
- Review of the management system (management review)

The principles of the performance of activities relating to monitoring and measurement are regulated by the following document as well [P. 40]. Monitoring and measurement are followed by the activities pertaining to analysis and settlement of identified non-conformities and potential non-conformities. The aforementioned regulatory document contains a list of the main variations of the internal as well as external monitoring and inspection instruments serving for the purpose of diagnostics of the systems, products and processes of ČEZ, a. s. In addition, it regulates the application of specific requirements imposed on monitoring and measurement by the defined relevant legal regulations and accepted requirements of the parties involved. Monitoring and measurement of the processes supporting safety, including the definition of the performance indicators is described in detail in the guide [P. 41] and document [P. 42].

Monitoring of the external partners satisfaction is based on the identification of external customers and parties involved, which is provided in the rule [P. 2]. Regulatory bodies are an important involved party. Among the measurable parameters of customer satisfaction, there are outputs of supervision (penalties, non-conformities arising from the supervision audits). Recapitulation of the used means of monitoring of the satisfaction of the aforementioned parties is provided in the document [P. 40].

Internal Control System

For the purpose of verification of the determined objectives on all levels of management, an internal control system was implemented in ČEZ, a. s. The internal control system has a cross-section character. It represents the necessary feedback in the management process and influences the decision-making process considerably by providing information. It represents all activities of the employees in managerial positions by means of which they discover whether the achieved results correspond with the planned ones. It assures self-assessment of the management functionality and effectiveness on all managerial levels.

The principle of the control system is methodical and periodic performance of comparison with predefined requirements, expectations and objectives, which are defined within sufficient scope and depth. Based on evaluation and analysis of the achieved results, or the analysis of data discovered within the framework of the controlling activities, objective conclusions are determined, which result in the proposals of effective measures aiming at remedy and corrective measures.

Independent Assessment and Supervision

Independent assessments, including analytical activities, are applied where required by the general binding legal regulations (impact on nuclear safety, radiation protection, technical safety, etc.) or where advisable. Among the methods of independent assessment, there are customer audits and the audits of quality, the environment and safety, supervision of the performance of activities, assessment by external experts (“peer”) and technical reviews.

Independent assessment and permanent supervision of the compliance with nuclear safety, radiation protection, physical protection, technical safety, occupational safety and health, environmental protection, and emergency preparedness in respect of activities important from the point of view of safety performed in nuclear power plants is executed by the Safety Department. Results of the independent assessment of the safety condition of the nuclear power plants are submitted to the Temelín NPP and Dukovany NPP management in the form of regular Safety Reports (monthly, annual) and important findings are submitted at the meetings of the Production Division, the Production Division Safety Committee and the Safety Committee of ČEZ, a.s. The independent assessment and feedback for strategic management in the area of safety is secured by the Safety Inspectorate Division of ČEZ Group.

Objectives of independent assessment are reviewed on a periodic basis to reflect issues that the management is currently facing and pending activities. Therefore, a combination of different types of assessment and verification is used to get the most balanced results.

The results of independent assessment are submitted to the strategic management for review and adoption of the necessary measures. The Board of the Directors of ČEZ, a. s., is submitted a regular summary report on the results of independent assessment by the employees of the Internal Audit Division – see the document [P. 23].

Results of independent assessment of the integrated management system are submitted to the relevant members of the strategic management through the Quality and Management System Division – see the guide [P. 4].

Safety Inspections and Feedback for Strategic Management

For the purpose of improvement of independent feedback for the strategic management within the framework of nuclear activities in ČEZ, a. s. (i.e. in the areas, which have influence on safe production of electric power and heat regulated by Act No. 18/1997 Coll. [L. 2]), there are inspectors appointed in the Safety Inspectorate Division of ČEZ Group performing the monitoring of nuclear activities. Monitoring of nuclear activities is performed in particular in the following areas:

- B01 - Technical safety
- B03 - Fire protection
- B04 - Nuclear safety
- B05 - Occupational safety and health protection
- B07 - Physical Protection of Nuclear Installations and Nuclear Materials
- B08 - Radiation protection
- B09 - Environmental protection
- B10 - Emergency preparedness
- B12 - Safety licence for the operation of NPP

Review of the Management System

Review of the management system serves for the determination of suitability, relevance and effectiveness of the management system for the achievement of the defined objectives. Within the framework of the process, the following is performed:

- Analyses and investigation of deviations of the management system against the requirements imposed by Decree No. 132/2008 Coll. [L. 258]
- Assessment of the management system, processes, documentation and organisational scheme against the requirements imposed by Decree No. 132/2008 Coll. [L. 258]
- Proposing improvement (remedial, corrective) measures to achieve the qualification and improvement of the management system
- Control system monitoring

In addition to measures implemented in the course of year, the results are summarized in the Annual Report on the Management System Review. The Report on Management System Review is prepared by the Quality Management Division as a basis for the review performed by the strategic management considering the defined policies and objectives, at regular intervals (at least once a year).

The Report on Nuclear Activities Monitoring considering nuclear safety, radiation protection, fire safety, physical protection, occupational safety and health protection is prepared by the Safety Inspectorate Division of ČEZ Group at regular intervals once every quarter of a year as a basis for the meeting/decision of the Board of Directors.

Outputs from the Management System Review

Based on results of the review (report on review), the strategic management decides on measures to be taken in relation to the following:

- Improvement of the effectiveness of the management system and the processes thereof and the need for implementation of changes in the management system
- Improvement of the product in relation to customer requirements
- Needs of resources
- Potential need to change the policies, objectives, target values or other element of the management system in accordance with the commitment to continual improvement

Non-Conformities, Corrective and Preventive Measures

Requirements concerning mechanisms to handle non-conformities found in the integrated management system are included in the document [P. 40]. The binding algorithm of handling of non-conformities is elaborated in detail in the document [P. 31].

The document also defines the general principles applicable to the classification of non-conformities relevance. Basic classification is into major and others. In the case of major non-conformities, the standard imposes the duty to handle thereof by means of a complete procedure, which includes the following:

- Registration and classification of the non-conformity

- Definition and implementation of immediate measures to limit the effect of the non-conformity duration
- Remedy of the non-conformity (recovery of the required parameters of the activity concerned)
- Analysis of the non-conformity causes
- Proposal and implementation of corrective measures to eliminate the non-conformity causes
- Identification of associated potential non-conformities
- Proposal and approval of preventive measures to eliminate potential non-conformities
- Monitoring of the implementation of the corrective and preventive measures
- Review of the effectiveness of the corrective and preventive measures

In the case of the other non-conformities, an abbreviated procedure completed with the non-conformity remedy is allowed.

The document aims to define the categorization of major non-conformities for graded handling of major non-conformities. For the category of especially major non-conformities, the division may define, in its control system, a standard-exceeding procedure for the handling of non-conformity consisting in, for example, compulsory identification of the non-conformity root causes.

Events occurring in the NPP are settled in accordance with the document [P. 43]. Registration and monitoring of the non-conformities settlement are supported by the compliant database applications.

Requirements imposed by the document [P. 40] are gradually applied in the safety-relevant divisions in their own control systems. In addition to recapitulation of control activities implemented in the division, they include a list of possible types of non-conformities, which might be identified by the control activity of the division, including the procedure for handling of these types of non-conformities.

In accordance with the document [P. 40], potential non-conformities are handled by analogous means and procedures as in the case of non-conformities.

Integrated Management System Improvement

Improvement of the integrated management system forms an integral part of the management system and is based on the assessment of the quality management system performed by the Quality and Management System Division with respect to the Quality Management Policy and the guide [P. 4]. Improvement is supported by the processes in the management area R04. Improvement is supported by the process of the handling of non-conformities (as well as deviations in the achievement of objectives).

The management area R04 defines processes, described in the guide [P. 4], with a view to influencing the management system of ČEZ, a. s., in a manner enabling to apply the means of modern quality management within the necessary scope and to integrate them in an effective and practical manner.

1.8 SUMMARY LIST OF DATA USED FOR THE PREPARATION OF THE INITIAL SAFETY ANALYSIS REPORT

L. 1 Vyhláška č. 215/1997 Sb., o kritériích na umísťování jaderných zařízení a velmi významných zdrojů ionizujícího záření, ve znění pozdějších předpisů (Decree No. 215/1997 Coll., on criteria for siting of nuclear installations and very significant ionising radiation sources, as amended)

L. 2 Zákon č. 18/1997 Sb., o mírovém využívání jaderné energie a ionizujícího záření (atomový zákon) a o změně a doplnění některých zákonů, ve znění pozdějších předpisů (Act No. 18/1997 Coll., on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act) and on Amendments and Additions to Related Acts)

L. 3 Vyhláška č. 423/2001 Sb., kterou se stanoví způsob a rozsah hodnocení přírodních léčivých zdrojů a zdrojů přírodních minerálních vod a další podrobnosti jejich využívání, požadavky na životní prostředí a vybavení přírodních léčebných lázní a náležitosti odborného posudku o využitelnosti přírodních léčivých zdrojů a klimatických podmínek k léčebným účelům, přírodní minerální vody k výrobě přírodních minerálních vod a o stavu životního prostředí přírodních léčebných lázní (vyhláška o zdrojích a lázních), ve znění pozdějších předpisů (Decree No. 423/2001 Coll., specifying the method and extent of the evaluation of natural medicinal sources and sources of natural mineral waters and other details concerning their use, requirements for the environment and equipment of natural spas and particulars of an expert opinion on usability of natural medicinal sources and climatic conditions for therapeutic purposes, natural mineral waters for the production of natural mineral waters and on the condition of the environment of natural spas (Regulation on Sources and Spas), as amended)

L. 4 Vyhláška č. 307/2002 Sb., o radiační ochraně, ve znění pozdějších předpisů (Decree No. 307/2002 Coll., on radiation protection, as amended)

L. 5 IAEA SF-1 Fundamental Safety Principles, Safety Fundamentals Series, Vienna 2006

L. 6 IAEA NS-R-3 Site Evaluation for Nuclear Instalation, Safety Requirements, Vienna 2003

L. 7 IAEA NS-G-1.5 External Events Excluding Earthquakes in the Design of Nuclear Power Plants, Safety Guide, Vienna, 2003

L. 8 NS-G-1.9 Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants, Safety Guide, Vienna, 2004

L. 9 IAEA NS-G-3.1 External Human Induced Events in Site Evaluation for Nuclear Power Plants, Safety Guide, Vienna, 2002

L. 10 IAEA NS-G-3.2 Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Safety Guide, Vienna, 2002

L. 11 IAEA NS-G-3.3 Evaluation of Seismic Hazards for Nuclear Power Plants NS-G-9 Seismic Hazards in Site Evaluation for Nuclear Installations Safety Guide, Vienna, 2003

L. 12 AEA NS-G-3.4 Meteorological Events in Site Evaluation for Nuclear Power Plants, Safety Guide, Vienna, 2004

L. 13 IAEA NS-G-3.6 Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants Safety Guide, Vienna 2004

L. 14 IAEA SSG-9 Seismic Hazards in Site Evaluiation for Nuclear Installations, Safety Guide, Vienna, 2010

L. 15 IAEA TECDOC-1341 Extreme External Events in the Design and Assessment of Nuclear Power Plants, Technical Document, Vienna, 2010

L. 16 IAEA TECDOC-343 Application of Microearthquake Surveys in Nuclear Power Plant Siting, Technical Document, Vienna, 1985

L. 17 IAEA No. 50-SG-S11A Extreme Meteorological Events in Nuclear Power Plant Siting, Excluding Tropical Cyclones, Safety Guides, Vienna, 1981

L. 18 Předprovozní bezpečnostní zpráva pro 1. a 2. blok JE Temelín (Preoperational safety report for Temelín ETE1,2 Units)

L. 19 IAEA NS-G-1.6 Seismic design and qualification for nuclear power plants, Safety Guide, Vienna, 2003

L. 20 Vnější havarijní plán JE Temelín, červen 2006 (Off-site Emergency Plan of Temelín NPP, June 2006)

L. 21 IAEA GS-R-2 Preparedness and Response for a Nuclear or Radiological Emergency Requirements, Vienna, 2002

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L. 286 Nařízení vlády č. 61/2003 Sb., o ukazatelích a hodnotách přípustného znečištění povrchových vod a odpadních vod, náležitostech povolení k vypouštění odpadních vod do vod povrchových a do kanalizací a o citlivých oblastech, ve znění nařízení vlády č. 229/2007 Sb. a nařízení vlády č. 23/2011 Sb. (Government Decree No. 61/2003 Coll., on indicators and values for admissible levels of surface and waste water pollution, on formalities concerning permission to release sewage into surface waters and into the sewer system, and on sensitive areas, as amended by Government Decree No. 229/2007 Coll. and Government Decree No. 23/2011 Coll.)

L. 287 Zákon č. 44/1988 Sb., o ochraně a využití nerostného bohatství (horní zákon), ve znění pozdějších předpisů (Act No. 44/1988 Coll., on Protection and Utilisation of Mineral Wealth (Mining Act), as amended)

P. 1 SKČ_PA_0001 - Systém řízení Skupiny ČEZ (Management System of ČEZ Group)

P. 2 ČEZ_PA_0022 - Manuál integrovaného systému řízení (Integrated Management System Manual)

P. 3 ČEZ_ST_0002 - Požadavky na integrovaný systém řízení (Requirements for Integrated Management System)

P. 4 ČEZ_SM_0121 - Management kvality (Quality Management)

P. 5 ČEZ_ST_0014 - Procesní řízení (Process Management)

P. 6 ČEZ_PA_0001 - Organizační řád ČEZ, a. s. (Organisational Rules of ČEZ, a. s.)

P. 7 ČEZ_PA_0017 - Organizační struktura, poslání a působnost útvarů (Organisational Structure, Mission and Field of Activity of Divisions)

P. 8 ČEZ_PR_1107 - Poslání útvarů D-2 liniové struktury útvaru VJE (Mission of Divisions of D-2 Line Structure of Nuclear Power Plant Construction Division)

P. 9 ČEZ_PMP_0001 - Projekt ETE 3,4 – Plán managementu projektu Dostavba Jaderné elektrárny Temelín (ETE3,4 Project – Project Management Plan of Completion of Temelín Nuclear Power Plant)

P. 10 ČEZ_PZJ_J63pol.0002_ETE - Program zabezpečení jakosti pro činnost umístění jaderného zařízení „3. a 4. blok“ v lokalitě JE Temelín (Quality Assurance Programme for the Activity Related to Siting of the Nuclear Installation "Units 3 and 4" at Temelín NPP)

P. 11 SKČ_PA_0006 – Řízení bezpečnosti ve Skupině ČEZ (Safety Management in ČEZ Group)

P. 12 ČEZ_ME_0758 - Tvorba programu zabezpečování jakosti programu jakosti změny/rekonstrukce (Creation of the Quality Assurance Programme for the Change/Reconstruction Quality Programme)

P. 13 ČEZ_ME_0824 - Projekt ETE 3,4 – Postup podle atomového zákona v etapě územního řízení (ETE3,4 – Procedure in Accordance with the Atomic Act in the Stage of Site Decision Proceedings)

P. 14 SKČ_SM_0023 – Nákup (Procurement)

P. 15 SKČ_PP_0081 - Řízení nákupní funkce (Purchasing Function Management)

P. 16 SKČ_PP_0077 - Obchodní zajištění nákupního požadavku (Commercial Provision for Purchase Request)

P. 17 ČEZ_PP_0362 - Hodnocení a kvalifikace dodavatelů (Supplier Evaluation and Qualification)

P. 18 ČEZ_ME_0747 - Vstupní posouzení dodavatele pro JE (Initial Assessment of Supplier for NPP)

P. 19 ČEZ_ME_0753 - Zadávání veřejných zakázek (Public Procurement)

P. 20 ČEZ_ME_0639 - Postup při kontraktaci (Contracting Procedure)

P. 21 ČEZ_PP_0274 - Audity kvality, EMS a BOZP (Audits of Quality, Environment Management System and Occupational Safety and Health)

P. 22 ČEZ_ME_0772 - Plánování a provádění auditů kvality, EMS a BOZP (Planning and Performing Audits of Quality, Environment Management System and Occupational Safety and Health)

P. 23 SKČ_PP_0050 - Zásady provádění interního auditu ve Skupině ČEZ (Principles of Internal Audit in ČEZ Group)

P. 24 ČEZ_PP_0111 - Zákaznické audity dodavatelů (Customer Audits of Suppliers)

P. 25 ČEZ_SM_0134 – Řízení projektu (Project Management)

P. 26 ČEZ_PP_0342 - Výstavba a komplexní obnova elektráren v ČR a zahraničí (Construction and Comprehensive Reconstruction of Power Plants in the Czech Republic and Abroad)

P. 27 ČEZ_ME_0827 - Projekt ETE 3,4 – Posuzování vlivů na životní prostředí (EIA) (ETE3,4 Project – Environmental Impact Assessment (EIA))

P. 28 SKČ_PP_0079 - Správa dokumentů (Document Administration)

P. 29 SKČ_SM_0016 - Spisový a skartační řád Skupiny ČEZ (Records And Shredding Regulations of ČEZ Group)

P. 30 ČEZ_ME_0843 - Projekt ETE 3,4 – Ukládání dokumentů (ETE3,4 Project – Document Storage)

P. 31 ČEZ_PP_0360 - Vypořádání neshod systému řízení (Handling of Non-Conformities of the Management System)

P. 32 ČEZ_SM_0142 - Rozvoj LZ (Human Resources Development)

P. 33 ČEZ_ME_0626 - Specifikace pro požadování položek (Specification for Items Requesting)

P. 34 ČEZ_ME_0745 - Zadávání neveřejných zakázek (Non-Public Procurement)

P. 35 ČEZ_SM_0117 - Řízení bezpečnosti v ČEZ, a. s. (Safety Management in ČEZ, a. s.)

P. 36 ČEZ_PR_1038 - Personální politika ČEZ, a. s. (Human Resources Policy of ČEZ, a. s.)

P. 37 ČEZ_SM_0141 - Zajišťování LZ (Provision of Human Resources)

P. 38 ČEZ_PP_0352 - Řízení znalostí (Knowledge Management)

P. 39 ČEZ_SM_0129 - Optimalizace požadavků na dodavatelský systém (Optimisation of Requirements for Supplier System)

P. 40 ČEZ_ST_0034 - Kontrolní systém (Control System)

P. 41 ČEZ_SM_0109 - Jaderná bezpečnost (Nuclear Safety)

P. 42 ČEZ_PP_0258 - Bezpečnostní reporting (Safety Reporting)

P. 43 ČEZ_PP_0205 - Zpětná vazba z provozních zkušeností (Operational Experience Feedback)

P. 44 SKČ_PP_0054 - Zajištění způsobilosti zaměstnanců (Personnel Competency)

P. 45 ČEZ_ME_0903 - Průzkum Kultury bezpečnosti ve Skupině ČEZ (Examination of Safety Culture in ČEZ Group)

P. 46 ČEZ_PRGR_1203 - Výbor pro bezpečnost zařízení ČEZ, a. s. (Committee on Safety of Installations of ČEZ, a. s.)

P. 47 ČEZ_ST_0056 - Kultura bezpečnosti v segmentu divize výroba (Safety Culture in the Segment Production Division)

P. 48 ČEZ_ME_0203 - Program monitorování okolí ETE (Monitoring programme for the surroundings of the Temelín NPP)

P. 49 ČEZ_ME_0455 - Monitoring programme for discharges from the Temelín Nuclear Power Plant

P. 50 ČEZ_PRGR_1008 - Program monitorování výpustí z jaderné elektrárny Temelín (Safety and Environmental Protection Policy)

1.8.1 PUBLICLY AVAILABLE DATA

L. 1, L. 2, L. 3, L. 4, L. 5, L. 6, L. 7, L. 8, L. 9, L. 10, L. 11, L. 12, L. 13, L. 14, L. 15, L. 16, L. 17, L. 19, L. 20, L. 21, L. 22, L. 23, L. 24, L. 25, L. 26, L. 27, L. 28, L. 29, L. 30, L. 33, L. 46, L. 49, L. 50, L. 52, L. 53, L. 54, L. 55, L. 56, L. 58, L. 59, L. 60, L. 61, L. 62, L. 63, L. 64, L. 65, L. 66, L. 67, L. 68, L. 69, L. 70, L. 71, L. 72, L. 73, L. 74, L. 75, L. 76, L. 77, L. 78, L. 79, L. 80, L. 81, L. 82, L. 83, L. 84, L. 86, L. 87, L. 88, L. 90, L. 91, L. 93, L. 94, L. 95, L. 96, L. 97, L. 98, L. 99, L. 100, L. 101, L. 102, L. 104, L. 105, L. 106, L. 108, L. 109, L. 110, L. 111, L. 112, L. 114, L. 115, L. 116, L. 117, L. 118, L. 119, L. 120, L. 121, L. 122, L. 123, L. 124, L. 125, L. 126, L. 127, L. 128, L. 129, L. 130, L. 133, L. 134, L. 135, L. 136, L. 137, L. 138, L. 139, L. 140, L. 141, L. 142, L. 143, L. 144, L. 145, L. 146, L. 147, L. 148, L. 149, L. 150, L. 151, L. 152, L. 153, L. 154, L. 155, L. 156, L. 157, L. 165, L. 167, L. 169, L. 170, L. 171, L. 172, L. 174, L. 175, L. 176, L. 177, L. 178, L. 180, L. 181, L. 182, L. 183, L. 184, L. 185, L. 186, L. 187, L. 188, L. 189, L. 191, L. 193, L. 194, L. 195, L. 196, L. 197, L. 198, L. 199, L. 204, L. 205, L. 206, L. 207, L. 208, L. 209, L. 214, L. 215, L. 217, L. 218, L. 220, L. 225, L. 227, L. 228, L. 230, L. 231, L. 237, L. 243, L. 245, L. 247, L. 248, L. 250, L. 252, L. 254, L. 255, L. 256, L. 257, L. 258, L. 259, L. 260, L. 261, L. 262, L. 263, L. 264, L. 266, L. 267, L. 268, L. 269, L. 270, L. 271, L. 272, L. 273, L. 274, L. 275, L. 276, L. 277, L. 278, L. 279, L. 280, L. 281, L. 282, L. 283, L. 284, L. 285, L. 286, L. 287

This data is not submitted with the Initial Safety Analysis Report to the Safety Office for Nuclear Safety due to its public availability (Internet, libraries, publications, standards, acts, decrees, etc.).

1.8.2 ADDITIONAL DATA

L. 36, L. 37, L. 38, L. 39, L. 40, L. 41, L. 42, L. 51, L. 57, L. 89, L. 92, L. 103, L. 107, L. 113, L. 166, L. 173, L. 210, L. 211, L. 221, L. 223, L. 224, L. 246

This data is owned by ČEZ, a. s., and is not classified as a trade secret in accordance with Act No. 513/1991 Coll. This data is submitted together with the Initial Safety Analysis Report to the State Office for Nuclear Safety.

1.8.3 ADDITIONAL DATA CLASSIFIED AS A TRADE SECRET IN ACCORDANCE WITH ACT NO. 513/1991 COLL.

L. 18, L. 31, L. 32, L. 34, L. 35, L. 43, L. 44, L. 47, L. 48, L. 85, L. 131, L. 132, L. 158, L. 159, L. 160, L. 161, L. 162, L. 163, L. 164, L. 168, L. 179, L. 190, L. 192, L. 202, L. 203, L. 212, L. 213, L. 219, L. 222, L. 226, L. 229, L. 232, L. 233, L. 234, L. 235, L. 236, L. 244, L. 249, L. 253, L. 265, L. 279

This data is owned by ČEZ, a. s., and is classified as a trade secret in accordance with Act No. 513/1991 Coll. This data is submitted together with the Initial Safety Analysis Report to the State Office for Nuclear Safety.

1.8.4 INTERNAL CONTROL DOCUMENTS OF ČEZ, A. S.

P. 1 – P. 50

Data that is available to the State Office for Nuclear Safety and is not of a public nature. This data is not submitted together with the Initial Safety Analysis Report to the State Office for Nuclear Safety.

1.8.5 REQUESTED CORRESPONDENCE OF ČEZ, A. S.

L. 45, L. 200, L. 201, L. 216, L. 238, L. 239, L. 240, L. 241, L. 242, L. 251

This correspondence was requested by ČEZ, a. s., for the purposes of preparing the Initial Safety Analysis Report. This data is submitted together with the Initial Safety Analysis Report to the State Office for Nuclear Safety.

1.9 ANNEXES TO THE INITIAL SAFETY ANALYSIS REPORT

Dwg. 1 Vicinity Plan, Scale 1:30,000, Arch. No. 5093-D-111671

Dwg. 2 Surface Streams and Water Bodies, Scale 1:10,000, Arch. No. 5093-D-111669

Dwg. 3 Layout of the Location of Sources of External Influences, Scale 1: 25,000, Arch. No. 5093-D-111672

Dwg. 4 Layout of the Location of Sources of Internal Influences, Scale 1:5,000, Arch. No. 5093-D-111673

Dwg. 5 Structures of Interest in the Layout Plan, Scale 1:100,000, Arch. No. 5093-D-111670

Annex 1 Delineation of the Area for Planned Completion of Temelín NPP in Power Plant Master Plan

Annex 2 Vicinity Plan

Annex 3 Possible Layout for the Completion of Two Units on the Premises of Temelín NPP

Annex 4 Basic Organisational Chart of the ČEZ joint-stock company as at 1 January 2012

Annex 5 General Management Model of ČEZ, a. s.

2 DESCRIPTION AND EVIDENCE OF SUITABILITY OF THE SITE FROM THE ASPECT OF SITING CRITERIA FOR NUCLEAR INSTALLATIONS

In the 1980s, the site of the Temelín Nuclear Power Plant was prepared for siting of four VVER1000 units with unit power 1,000 MW. The limitation of the construction of the power plant to two units in 1992 gave rise to the formation of an area usable for the future construction. The plan to construct another two nuclear units in the above mentioned area requires a permitting process for their siting and construction to the full extent, in accordance with the current legal and technical regulations.

The subject of this chapter is to summarize the results of the assessment of suitability of the Temelín site for siting of another two units of the nuclear power plant. The site was assessed and this report was arranged in accordance with the expected requirements for the preparation of the Chapter "Description and Evidence of Suitability of the Site from the Aspect of Siting Criteria for Nuclear Installations" in the Initial Safety Analysis Report Temelín ETE3,4 in accordance with Appendix A to Act No. 18/1997 Coll. [L. 2].

Chapter 2 of this report is composed of eight subsections containing description and analysis of site characteristics important to siting of a nuclear installation (2.1 to 2.8), subsection 2.9 containing conclusions of the assessment of the Temelín site from the aspect of siting criteria for a nuclear installation in accordance with Decree No. 215/1997 Coll., [L. 1] and subsection 2.10 containing an overview of design measures necessary to ensure compliance of the plan for siting of the nuclear installation in the Temelín site with the conditional and exclusion criteria in accordance with Section 5 of Decree No. 215/1997 Coll. [L. 1].

The results of the surveys and analyses of the Temelín construction site carried out over the past 30 years for the original plan for the construction and operation of two implemented units are available for the assessment of siting of the new designed Temelín NPP Units 3 and 4. However, in accordance with Section 6 of Decree No. 215/1997 Coll. [L. 1], it was necessary to assess whether or not the original data over time had not become worthless.

The above-cited siting criteria for nuclear situations in accordance with Decree No. 215/1997 Coll., [L. 1] were used for the assessment of suitability of the site, which were applied through the provisions of the Safety Guide of the State Office for Nuclear Safety BN-JB-1.14 [L. 268] and the requirements for the site for siting a nuclear installation defined in the IAEA standard NS-R-3 [L. 6]. Other IAEA standards relating to siting of nuclear installations (see list of used documents [L. 7] to [L. 17] and [L. 21] to [L. 24]) were also used for the assessment of site characteristics. It was not necessary to include in the list of used IAEA standards specific standards relating to floods because the decision on the resistance of the construction site to flood in the Temelín near region could be made on the basis of assessment of the surface relief of the earth and the fact that the structures of the power plant situated on the bank of the Vltava River used for makeup water withdrawal and waste water disposal do not provide the functions necessary from the aspect of nuclear safety can be designed as standard industrial facilities and their flooding is supposed. The validity of this assessment was acknowledged by an independently ongoing assessment process within the stress tests in response to the event that occurred at Fukushima NPP.

The criteria[2] for assessing the suitability of the site for siting of a nuclear installation defined in Decree No. 215/1997 Coll. [L. 1], and the requirements for this site stipulated in the IAEA standard NS-R-3 [L. 6] are dealt with by individual paragraphs or groups of paragraphs addressing the same problems in Sections 2.1.7, 2.2.7, 2.3.7, 2.4.7, 2.5.7, 2.6.7, 2.7.7 and 2.8.7 of this Initial Safety Analysis Report. The complete wording of Sections 4 and 5 of Decree No. 215/1997 Coll. [L. 1], i.e. all criteria defined in this decree, was included in the assessment in the specific parts of the Initial Safety Analysis Report. The requirements stipulated in Sections 3, 4 and 5[3] in the IAEA standard NS-R-3 [L. 6] were applied to the assessment.

An overview of the findings of site assessment from the aspect of the criteria for assessing the suitability of the site for siting of the nuclear power plant as defined in Decree No. 215/1997 Coll. [L. 1], is given in Section 2.10 of this Initial Safety Analysis Report. The text of Section 2.9 includes references to descriptions and analyses presented in other parts of the Initial Safety Analysis Report to eliminate duplicities in the description. During the analysis of the Temelín site in accordance with Decree No. 215/1997 Coll. [L. 1], the site characteristics were analysed for full compliance with the siting requirements defined in Section 4. During the analysis of the site characteristics in accordance with the criteria under Section 5 of the aforementioned decree, the measures were determined in the event of deviations of the site characteristics from the criteria requirements to solve the unfavourable properties of the site in the prepared design of the power plant.

The above-mentioned facts show that Section 1 of the IAEA standard NS-R-3 [L. 6], which defines the content and specialization of the standard itself, was not directly applied by individual paragraphs in analysing the site characteristics. The requirements stipulated in Section 2 of the IAEA standard NS-R-3 [L. 6] are formulated as principles and general requirements, which are specified in following Sections 3, 4, 5 and 6 of the standard. Site assessment in accordance with the specific requirements stipulated in Sections 3, 4 and 5 of the IAEA standard NS-R-3 [L. 6] is the subject of Chapter 2, the requirements stipulated in Section 6 of the IAEA standard NS-R-3 [L. 6] for quality assurance are the subject of Chapter 6 hereof.

The interconnection between the individual general requirements of Section 2 of the IAEA standard NS-R-3 [L. 6] and the specific requirements stipulated in Sections 3, 4 and 5 of the IAEA standard NS-R-3 [L. 6] and analysed in this Initial Safety Analysis Report is described in Tab. 3. Whereas the general requirements defined in Section 2 of the IAEA standard NS-R-3 [L. 6] are applied to a various extent in more specific requirements of the IAEA standard cited, Tab. 3 indicates only decisive relations.

Similarly as the aforementioned Section 2 of the IAEA standard NS-R-3 [L. 6], the provisions of para. 3.55[4] are applied in other parts of the standard in assessing the

[2] The criteria are structured in items by sections and paragraphs marked with letters identical with their marking in Decree No. 215/1997 Coll. [L. 1]

[3] The requirements not related to the inland Temelín site are evaluated by reference to site type difference.

[4] Para. 3.55 of the IAEA requirements is worded as follows: "If the hazards for the nuclear installation are unacceptable and no practicable solution is available, the site shall be deemed unsuitable for siting of the nuclear installation."

site characteristics. The IAEA standard NS-R-3 [L. 6] determines the way the findings of the assessment of characteristics of the site for siting of a nuclear installation may be used. If any risk arising from a certain characteristic of the construction site is found to be unacceptable and the available measures for its reduction cannot be implemented, the construction site shall be evaluated as unsuitable. The aforementioned paragraph of the IAEA standard NS-R-3 [L. 6] also shows that if there is any measure available to reduce the risk arising from the site characteristics, the construction site shall be deemed suitable. From the aspect of the design of the nuclear installation at the considered construction site, these measures become a part of its design basis defined in Section 2.10.

Tab. 3 Overview of requirements in Section 2 of the IAEA standard NS-R-3 and their application to specific requirements

Paragraph in Section 2	Requirements in Section 2 of the IAEA standard NS-R-3	Related specific requirements in the IAEA standard NS-R-3
2.1.a)	In the evaluation of the suitability of a site for a nuclear installation, the aspects such as the effects of external events that may occur in the region of the particular site (these events could be of natural origin or human induced) shall be considered.	Section 3
2.1.b)	In the evaluation of the suitability of a site for a nuclear, the aspects such as the characteristics of the site and its environment that could influence the transfer to persons and the environment of radioactive material that has been released shall be considered.	Para. 4.1 to 4.9 and 4.14
2.1.c)	In the evaluation of the suitability of a site for a nuclear, the aspects such as the population density and population distribution and other characteristics of the external zone in so far as they may affect the possibility of implementing emergency measures and the need to evaluate the risks to individuals and the population shall be considered.	Para. 4.10 to 4.13
2.2	If the site evaluation for the three aspects cited indicates that the site is unacceptable and the deficiencies cannot be compensated for by means of design features, measures for site protection or administrative procedures, the site shall be deemed unsuitable.	All requirements in Sections 3, 4 and 5
2.3	In addition to providing the technical basis for the safety analysis report to be submitted to the nuclear regulatory body, the Technical Information obtained for use in complying with these safety requirements will also be useful in fulfilling the requirements for the environmental impact assessment for radiological hazards.	*)
	*) The technical basis for the safety analysis report and for the environmental impact assessment shall be coordinated so that the initial assumptions used for the assessment in the environmental impact assessment and the Initial Safety Analysis Report are identical.	
2.4	Site characteristics that may affect the safety of the nuclear installation shall be investigated and assessed.	All requirements defined in Sections 3, 4 and 5.

Paragraph in Section 2	Requirements in Section 2 of the IAEA standard NS-R-3	Related specific requirements in the IAEA standard NS-R-3
2.5	Proposed sites for nuclear installations shall be examined with regard to the frequency and severity of external natural and human induced vents and phenomena that could affect the safety of the installation.	All requirements defined in Section 3. The external natural and human induced events are evaluated with regard to the effect and, if cannot be neglected, with regard to the frequency of their possible occurrence.
2.6	The foreseeable evolution of natural and human made factors in the region that may have a bearing on safety shall be evaluated for a time period that encompasses the projected lifetime of the nuclear installation. These factors, particularly population growth and population distribution, shall be monitored over the lifetime of the nuclear installation. If necessary, appropriate measures shall be taken to ensure that the overall risk remains acceptably low. There are the following means available to ensure that risks are acceptably low: design features, measures for site protection (e.g. dykes for flood control) and administrative procedures. Design features and protective measures are the preferred means of ensuring that risks are kept acceptably low.	Para. 4.11
2.7	The hazards associated with external events that are to be considered in the design of the nuclear installation shall be determined. For an external event. (or a combination of events) the parameters and the values of those parameters that are used to characterize the hazards should be chosen so that they can be used easily in the design of the installation.	All requirements defined in Section 3. If any requirements for the design arise from the assessment of the individual influences, these impacts on the design are defined in Section 2.10 hereof, including the values of the parameters of such influences.
2.8	In the derivation of the hazards associated with external events, consideration should be given to the effects of the combination of these hazards with the ambient conditions (e.g. hydrological, hydrogeological and meteorological conditions).	All requirements defined in Section 3.
2.9	In the analysis to determine the suitability of the site, consideration shall be given to additional matters relating to safety such as the storage and transport of input and output materials (uranium ore, UF_6, UO_2, etc.), fresh and spent fuel and radioactive wastes.	Requirements defined in Sections 3 and 4.
2.10	The possible non-radiological impact of the installation, due to chemical or thermal releases, and the potential for explosion and the dispersion of chemical products shall be taken into account in the site evaluation process.	Para. 3.49 and 3.51.
2.11	The potential for interactions between nuclear and non-nuclear effluents, such as the combination of heat or chemicals with radioactive material in liquid effluents, should be considered.	Para. 4.4 to 4.6
2.12	For each proposed site the potential radiological impacts in operational	Para. 4.14 and

Paragraph in Section 2	Requirements in Section 2 of the IAEA standard NS-R-3	Related specific requirements in the IAEA standard NS-R-3
	states and in accident conditions on people in the region, including impacts that could lead to emergency measures, shall be evaluated with due consideration of the relevant factors, including population distribution, dietary habits, use of land and water, and the radiological impacts of any other releases of radioactive material in the region.	Chapter 4 of the Initial Safety Analysis Report.
2.13	For nuclear power plants, the total nuclear capacity to be installed on the site should be determined as far as possible at the first stages of the siting process. If it is proposed that the installed nuclear capacity be significantly increased to a level greater than that previously determined to be acceptable, the suitability of the site shall be re-evaluated, as appropriate.	**)
	**) The site assessment reflects the expected capacity of the new units as well as the capacity of the existing units.	
2.14	Proposed sites shall be adequately investigated with regard to all the site characteristics that could be significant to safety in external natural and human induced events.	All requirements defined in Section 3. The scope of site investigations shall meet the legislative requirements and usual international practice. The scope of investigations is indicated in the individual sections related to specific influences.
2.15	Possible natural phenomena and human induced situations and activities in the region of a proposed site shall be identified and evaluated according to their significance for the safe operation of the nuclear installation. This evaluation should be used to identify the important natural phenomena or human induced situations and activities in association with which potential hazards are to be investigated.	All requirements defined in Section 3. The external influences in accordance with the requirements of the Czech legislation and in accordance with the requirements stipulated in the IAEA standard NS-R-3 [L. 6] were evaluated in terms of their significance for the safe operation of the nuclear installation. The impacts on the design are specified in Section 2.10.
2.16	Foreseeable significant changes in land use shall be considered, such as the expansion of existing installations and human activities or the construction of high risk installations.	Para. 3.44, 4.11.

Paragraph in Section 2	Requirements in Section 2 of the IAEA standard NS-R-3	Related specific requirements in the IAEA standard NS-R-3
2.17	Prehistorical, historical and instrumentally recorded information and records, as applicable, of the occurrences and severity of important natural phenomena or human induced situations and activities shall be collected for the region and shall be carefully analysed for reliability,accuracy and completeness.	Para. 3.2, 3.8, 3.12, 3.18, 3.19, and 3.52 .
2.18	Appropriate methods shall be adopted for establishing the hazards that are associated with major external phenomena. The methods shall be justified in terms of being up to date and compatible with the characteristics of the region. Special consideration should be given to applicable Probabilistic methodologies. It should be noted that probabilistic hazard curves are generally needed to conduct probabilistic safety assessments for external events.	Para. 3.4, 3.10, 3.19, 3.45, 3.51, and 4.3.
2.19	The size of the region to which a method for establishing the hazards associated with major external phenomena is to be applied shall be large enough to include all the features and areas that could be of significance in the determination of the natural and human induced phenomena under consideration and for the characteristics of the event.	Para. 3.1, 3.5, 3.18.
2.20	Major natural and human induced phenomena shall be expressed in terms that can be used as input for deriving the hazards associated with the nuclear installation; that is, appropriate parameters for describing the hazard should be selected or developed.	All requirements defined in Section 3.
2.21	In the determination of hazards, site specific data shall be used, unless such data are unobtainable. In this case, data from other regions that are sufficiently relevant to the region of interest may be used in the Determination of hazards. Appropriate and acceptable simulation techniques may also be used. In general, data obtained for similar regions and simulation techniques may also be used to augment the site specific data.	All requirements defined in Section 3.
2.22	In the evaluation of a site to determine its potential radiological impact on the region for operational states and accident conditions that could lead to emergency measures, appropriate estimates shall be made of expected or potential releases of radioactive material, with account taken of the design of the installation and its safety features. These estimates shall be confirmed when the design and its safety features have been confirmed.	This is dealt with in Chapter 4 of the Initial Safety Analysis Report ETE3,4 and subsequently in analyses of the design used in ETE3,4.
2.23	The direct and indirect pathways by which radioactive material released from the nuclear installation could potentially reach and affect people and the environment shall be identified and evaluated.	Para. 4.1, 4.2, 4.4 and 4.5.
2.24	The site and the design for the nuclear installation shall be examined in conjunction to ensure that the radiological risk to the public and the environment associated with radioactive releases is acceptably low.	Requirements defined in Sections 3, 4 and 5.

Paragraph in Section 2	Requirements in Section 2 of the IAEA standard NS-R-3	Related specific requirements in the IAEA standard NS-R-3
2.25	The design of the installation shall be such as to compensate for any unacceptable potential effects of the nuclear installation on the region, or otherwise the site shall be deemed unsuitable.	***)
	***) If any impacts on the design arise from the evaluation of external hazards, measures to compensate for any hazards shall be included in the design. These impacts are summarized in Section 2.10 of this Initial Safety Analysis Report; in any case, these are typical technical solutions that ensure that the hazard remains at acceptable level.	
2.26	The proposed region shall be studied to evaluate the present and foreseeable future characteristics and the distribution of the population of the region. Such a study shall include the evaluation of present and future uses of land and water in the region and account shall be taken of any special characteristics that may affect the potential consequences of radioactive releases for individuals and the population as a whole.	Para. 4.4 to 4.15.
2.27. a)	In relation to the characteristics and distribution of the population, the combined effects of the site and the installation shall be such that: (a) For operational states of the installation the radiological exposure of the population remains as low as reasonably achievable and in any case is in compliance with national requirements, with account taken of international recommendations.	Para. 4.6, 4.9 and 4.13.
2.27. b)	(b) The radiological risk to the population associated with accident conditions, including those that could lead to emergency measures being taken, is acceptably low.	Para. 4.6, 4.9 and 4.13.
2.28	If, after thorough evaluation, it is shown that no appropriate measures can be developed to meet the above mentioned requirements, the site shall be deemed unsuitable for the location of a nuclear installation of the type proposed.	This requirement is not further detailed in the standard. It is equivalent to the principle of site assessment in accordance with Decree No. 215/1997 Coll. [L. 1]
2.29	The external zone for a proposed site shall be established with account taken of the potential for radiological consequences for people and the feasibility of implementing emergency plans, and of any External events or phenomena that may hinder their implementation. Before construction of the plant is started, it shall be confirmed that there will be no insurmountable difficulties in establishing an emergency plan for the external zone before the start of operation of the plant.	This requirement is not further detailed in the standard. It is equivalent to the principle of site assessment under Section 4(b) of Decree No. 215/1997 Coll. [L. 1]

On the basis of the assessment of site characteristics in accordance with the criteria defined in Section 5 of Decree No. 215/1997 Coll. [L. 1], the technically feasible requirements for the design of ETE3,4 were laid down, the implementation of which shall ensure compliance of the ETE3,4 site with the conditional criteria of Section 5 cited. The requirements for the design allowing to meet the requirements in Section 5 of Decree No. 215/1997 Coll. [L. 1], and the design parameters arising from external influences are defined in Section 2.10 hereof.

The characteristics of the Temelín site in the scope described in this Section shall be periodically verified, according to the results of investigations and as needed, amended [5] and the draft measures, including their reflection in the design of the installation, shall be modified, as needed.

[5] This involves the application of requirements stipulated in para. 5.1 of the IAEA standard NS-R-3 [L. 6]: "The characteristics of the natural and human induced hazards as well as the demographic, meteorological and hydrological conditions of relevance to the nuclear installation shall be monitored over the lifetime of the power plant. This monitoring shall be commenced no later than at the start of construction and shall be continued up until abandonment of the power plant. All the hazards and conditions that are significant for the licensing and safe operation of the installation shall be monitored."

2.1 GENERAL DATA ABOUT THE SITE (GEOGRAPHY AND DEMOGRAPHY)

2.1.1 SCOPE OF THIS SECTION

This section contains site specification, including topography, demographic characteristics, use of the site in terms of industrial, agricultural, commercial, residential infrastructure, administrative centres, transport routes, or other facilities of the infrastructure (e.g. hospitals, police stations, fire brigades). In addition, protective zones of the power plant delimited in the site are listed.

This information associated with the assessment of potential influences of the site on operation of the power plant is taken from the set of information.

2.1.2 SUMMARY OF FACTS

2.1.2.1 GENERAL DATA ABOUT THE SITE

The Temelín Nuclear Power Plant (Temelín NPP) and the area of the projected completion are situated in the southern part of the Czech Republic, specifically in the following territorial units under the act on territorial division of the state:

Tab. 4 ETE3,4 site from the administrative point of view

Region	District	Municipality with Extended Powers (MEP)	Municipality	Cadastre
South Bohemia	České Budějovice	Týn nad Vltavou	Temelín	Křtěnov (cadastre code 613975) – ETE1,2, ETE3,4 Březí u Týna nad Vltavou (cadastre code 613941) – ETE1,2 Temelínec (cadastre code 765813) – ETE1,2, ETE3,4

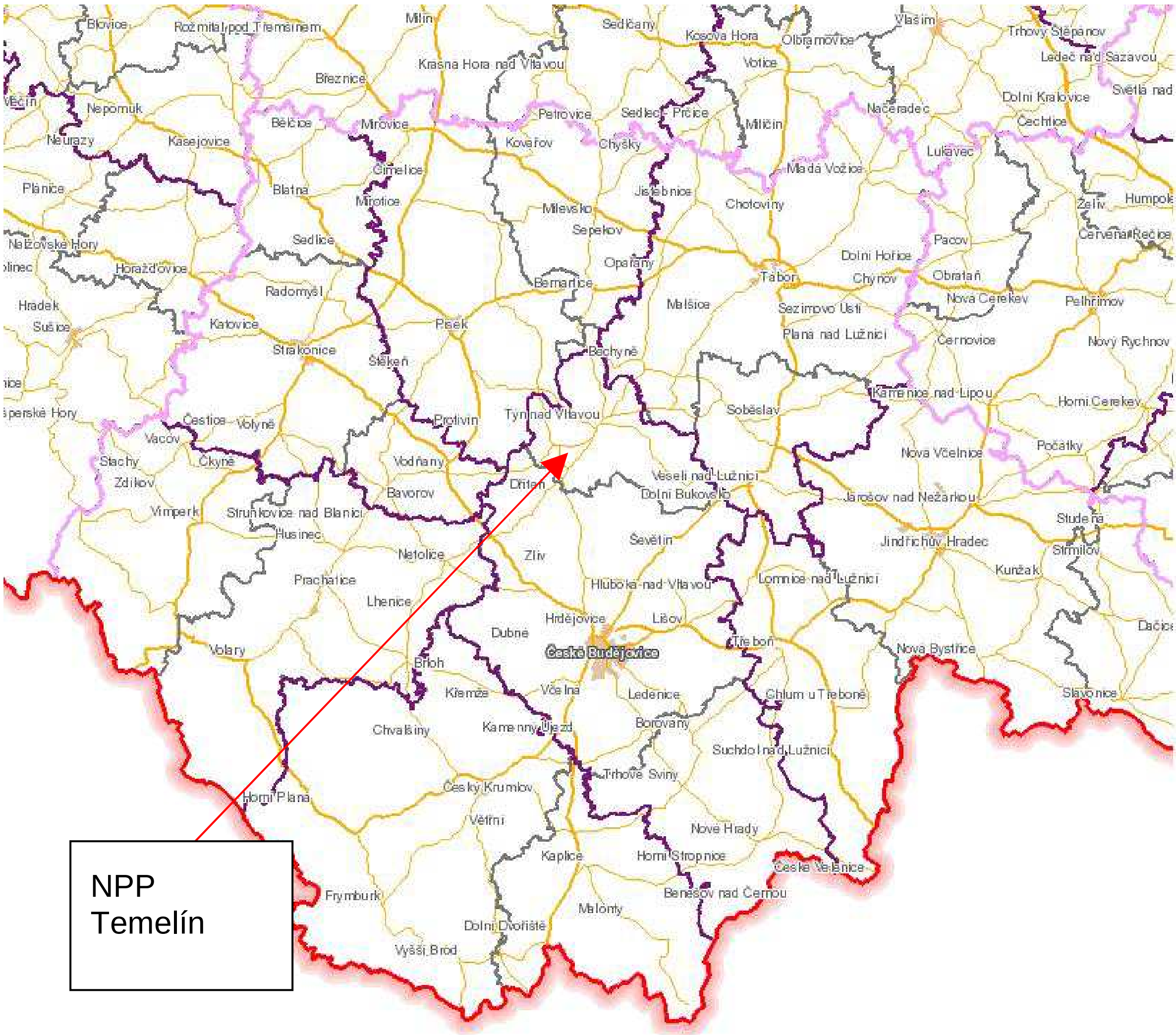

Fig. 2 Territorial division of the South Bohemian Region (region – districts – municipalities with extended powers)

Geographic coordinates of ETE3,4 are: approximately 49°11´north latitude and 14°22´ east longitude. Districts neighbouring to the district with the Temelín site are as follows:

- Český Krumlov
- Jindřichův Hradec
- Prachatice
- Písek
- Strakonice
- Tábor

The location of the construction site with regard to the neighbouring states and significant towns is shown in Fig. 3.

The area of ETE3,4 and the existing premises of ETE1,2 in a broader territorial context are shown in an Annex in Dwg. 1.

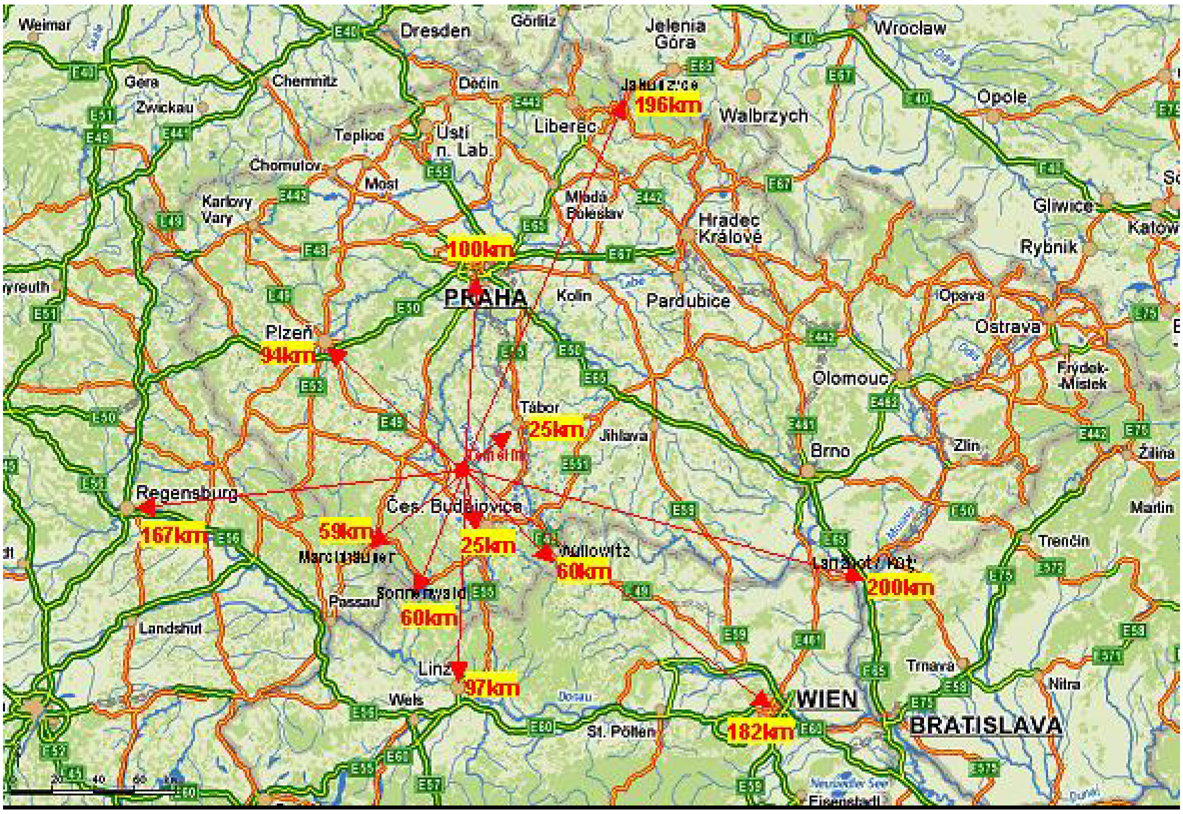

Fig. 3 Location of the construction site with regard to the neighbouring states and significant towns

The town and country planning in the area in question is based on the Development Principles of the South Bohemian Region approved by the South Bohemian Regional Assembly on 13 September 2011 [L. 218]. The plans of the individual municipalities follow the aforementioned Development Principles of the South Bohemian Region. Other information on the plan is provided in Section 2.1.2.4 hereof.

2.1.2.2 GEOGRAPHICAL DATA

2.1.2.2.1 Summary Description

Due to its geographical position and natural conditions, South Bohemia, or the South Bohemian Region, which the Temelín near region comes under, belongs to strategically significant territories, where the first human settlements occurred in the distant past. The South Bohemian Region is traditionally seen as an agricultural region with predominant fishpond industry and forestry. However, industry keeps developing there.

In terms of current development of the region, the territorial neighbourhood with the EU countries (the Federal Republic of Germany and Austria) plays an indispensable role, which is currently reflected in the deepening economic and culturally social relationships with both neighbouring states.

In addition to the above mentioned basic characteristics, the territory of South Bohemia is intensively used for recreational purposes and an effort to preserve the natural environment is mainly reflected in the southern part of the region through the establishment of the Šumava National Park and other protected areas (Třeboň region, Blansko Forest, etc.).

The site of the existing ETE1,2 with the total area approximately 135 ha is situated in the cadastral area of Křtěnov, cadastral area of Březí u Týna nad Vltavou and cadastral area of Temelínec. The area is situated on the top of a hill, which was, due to construction, levelled up to two basic altitudes, 503.00 m above sea level and 507.00 m above sea level. The existing structures important to nuclear safety are concentrated in the central part of the premises at the elevation of 507.00 m above sea level. This area is delimited by a triangle, when it is bounded on its south-east side by the road II/105 from Týn nad Vltavou to České Budějovice, on its south-west side by the road II/138 from the Temelín municipality and on its north side by 4 gas pipeline routes followed by the road II/141 from Temelín to Týn nad Vltavou. The area for the construction of ETE3,4 is situated west of ETE1,2. The total area for construction amounts to 84.93 ha and the area for site facilities amounts to 63.82 ha.

2.1.2.2.2 Meteorology and Hydrography

The power plant is situated in the Atlantic-continental area of the temperate climate zone of the northern hemisphere. Synoptic situations of west directions prevail in the area (39.9%). The frequency of the situation of east directions amounts to 15.7%, north situations 16.0% and south situations 7.5%. Air masses of oceanic and continental origin, which are mainly formed in middle latitudes, change here all year round. Bursts of air masses of tropical and arctic origin are frequent. The proportion of the oceanic and continental climate is balanced. The average annual temperature at the site for the period 1876–2009 is 8.4°C with a root mean square deviation of 0.8°C, the average temperature in the coldest month (January, -1.2°C), and the average temperature in the hottest month (June, 18.2°C). The average annual precipitation total for the period 1876–2009 is 591 mm with a root mean square deviation of 143 mm and ranges from 370 to 1,060 mm. (for details see [L. 211]).

The extreme and exceptional meteorological conditions occurring at the site are described in Section 2.4.2 hereof.

The site is not limited by increased occurrence of excessively unfavourable dispersion conditions in the atmosphere.

From a hydrographic point of view, the territory of the South Bohemian Region belongs to the basin of the upper and middle Vltava River with the affluents of the Otava, Lužnice, Malše, Blanice and many more rivers. More than 7,000 ponds have been built here in the past; their total acreage now amounts to more than 30,000 hectares. The large water dam of Lipno (the biggest water surface in the Czech Republic measuring 4,870 ha), the water dam of Orlík surrounded by large recreational areas and the water dam of Římov, which supplies a significant part of the South Bohemian Region with drinking water, have been built on the territory of the region. The water reservoir of Hněvkovice was built in connection with the construction of Temelín NPP (approximately 5 km from Temelín NPP) and the Kořensko plant for waste water disposal from Temelín NPP (approximately 7 km from Temelín NPP). Specific runoffs from the territory reach 3~5 $l/s.km^2$. The hydrology of the vicinity of Temelín NPP is described in Section 2.5.2 hereof.

2.1.2.2.3 Geomorphology

From the geomorphological point of view, the area of Temelín NPP is a part of the Bohemian-Moravian geomorphological system, the area of the Central Bohemian Uplands, the unit of the Tábor Uplands, the subunit of the Písek Uplands and the region of the Týn Uplands. The area of the Týn nad Vltavou municipality with extended powers, which includes Temelín NPP, forms a transition between the north edge of the Budějovice Basin and the northwest edge of the Třeboň Basin and uplands. The area of the Týn nad Vltavou municipality with extended powers at 400 to 500 m above sea level is of an undulating nature. The dominant shapes of the surface are the deeply incised valleys of the Vltava River, Lužnice River and some of their affluents. The lowest point of the subject area is situated in the valley of the Vltava River (approximately 350 m above sea level), the highest point is the Velký Kamýk (628 m above sea level) in the v Mehelnice Highlands approximately 7 km northwest of Temelín NPP. The area of ETE3,4 is situated at 503 to 507 m above sea level.

2.1.2.2.4 Environment

In accordance with Act No. 100/2001 Coll. [L. 255], the affected area from the environmental point of view shall mean an area "the environment and population of which could be significantly affected by the implementation of the plan". According to this definition, the affected area in the EIA documentation for ETE3,4 [L. 29] is limited to the area of the plan and its neighbourhood. In accordance with the above mentioned EIA documentation, the affected area is not located in the area with a special mode of nature conservation and landscape protection. Technically, this means:

- There is no specially protected area in the affected area within the meaning of Section 14 of Act No. 114/1992 Coll., on the Conservation of Nature and Landscape, nor is this affected area a part of a specially protected area. The affected area is not located in a national park or a protected landscape area, there are no national nature reserves, nature reserves, national nature monuments or nature monuments declared in the affected area.
- The affected area (the area of intended construction) does not include any features of the territorial system of ecological stability; the features of the territorial system of ecological stability as well as significant landscape features are present in the neighbourhood.
- The affected area is not a part of a natural park.
- The affected area is not a part of the Natura 2000 network.

The affected area does not include a water resource protective zone in accordance with Act No. 254/2001 Coll. [L. 284]. The affected area is not located in a protected area of natural water accumulation (CHOPAV).

In the affected area, there were found no conflicts of interests with the active mineral deposits, protected deposit areas and mining areas, registered within the deposit protection maps.

2.1.2.2.5 Forests

The area of the Týn nad Vltavou municipality with extended powers is medium-forested with pines and spruces, river corridors are flanked by forest stands as well as affluent valleys; in highland positions, on inter-water ridges, more open farming landscape appears, which also includes siting of ETE3,4.

Larger continuous forest areas occur to the northwest of the NPP area (approximately 2.5 km) and to the east and northeast at (1 - 3 km). From forest stands near the Temelín NPP, a park surrounding the Vysoký hrádek (approximate acreage 400 x 400 m) and similarly large reclaimed areas north of the site area are situated at the northeast corner of the site area. The issues related to forests are further described in Section 2.2.2.1.

2.1.2.2.6 Industrial Production, Storage of Hazardous Substances

Facilities in the area with a radius of 10 km from Temelín NPP include small-scale auxiliary manufacturing in Týn nad Vltavou and Temelín, and a planned bio-ethanol production plant at Býšov. Most of these facilities employ up to 25 people. The biggest employer in the defined are is ČEZ, a. s., at the Temelín Nuclear Power Plant. Its construction, which started in 1980, created a lot of new jobs and registered an increase in the number of the population in the region, especially in Týn nad Vltavou, where a new housing estate was built for power plant personnel. The only other company employing more than 100 people within 10 km is MIKRONA HOLDING, s.r.o., weapon and ammunition production, with its registered office in Týn nad Vltavou (about 5 km from Temelín NPP). The existing larger-scale industries in the České Budějovice District are situated farther away. In the site vicinity zone in accordance with Decree No. 215/1997 Coll. [L. 1], i.e. an area up to 3 km away from the boundary of Temelín NPP, only the above-mentioned manufacturing activities are included in Temelín municipality and the planned bio-ethanol production plant at Býšov.

The potential sources of events within 10 km from Temelín NPP are present in the below listed points:

- Switchyard in Kočín
- Petrol stations in Temelín and Týn nad Vltavou
- Wienerberger brickworks in Týn nad Vltavou
- Graphite plant in Týn nad Vltavou
- Slavětice stone quarry
- Bio-ethanol production plant at Býšov
- Forest stands in the vicinity of Temelín NPP
- Kočín building materials production plant[6]

The following stationary sources of internal events for ETE3,4 were identified at ETE1,2:

- Chemical warehouse
- Store of technical gases in cylinders
- Hydrogen warehouse management
- Diesel oil management
- Gas-burning boiler room
- Nitrogen warehouse management

[6] Whereas the Betonpres building materials production plant in Kočín reports no explosive and flammable materials, it is not further assessed hereinafter.

- Asphalt and bitumen radioactive waste storage facility at Active and Auxiliary Service Building for Primary Systems
- Gas extinguishing systems of Unit 2
- Oil systems of Unit 2
- Fuel system for diesel-generator stations west of Unit 2

The description of the sources of external events and the location of such sources of events are depicted in the enclosed drawings Dwg. 3 and Dwg. 4. The existence of these points is associated with the transport of hazardous materials by railway and by road. The topography of transport infrastructure is described in Section 2.1.2.2.9 hereof.

In accordance with the IAEA recommendation NS-G-3.1 [L. 9], the zone of surface source identification (SDV), as specified in Section 2.2.6, was delineated.

2.1.2.2.7 Agriculture

The Temelín near region is surrounded by farmed landscape. The following cooperative farms and other agricultural facilities can be found within a radius of 10 km from Temelín NPP:

- Žimutice Agricultural Cooperative (distance from Temelín NPP about 10 km)
- NOVÁ Dříteň Agricultural Cooperative (distance from Temelín NPP about 4 km)
- Private farmers
- Olešník Agricultural-Business Cooperative (distance from Temelín NPP about 8 km)
- Bohunice Large Scale Fattening Farm (distance from Temelín NPP about 3 km)
- Týn nad Vltavou, Číhovice Farm (distance from Temelín NPP about 4 km)

The above stated agricultural facilities do not include the registered quantity of substances dangerous to the proposed plan.

2.1.2.2.8 Energy Sources

The dominant energy source in the area within a radius of 10 km from ETE3,4 is ETE1,2 generating electric power in two generating units (each unit with VVER1000 nuclear installation with current achievable electrical output of 1,013 MWe). Other energy sources in the 10 km zone are: Hněvkovice Hydroelectric Power Plant (about 6 km north of the Temelín NPP) and Kořensko Hydroelectric Power Plant (about 4.5 km west of the Temelín NPP). Both power plants are included in the Vltava cascade. In the vicinity of the site according to Decree No. 215/1997 Coll. [L. 1], within a distance of 3 km from the border of ETE3,4, there are no other energy sources (except for ETE1,2) that would constitute a risk for the plan. During the construction, the premises of ETE3,4, situated adjacent to the premises of ETE1,2 will reach into the protection zone of ETE1,2 in a width of 20 m (see Section 47(7) of Act No. 458/2000 Coll. [L. 217]) from the fencing separating the guarded area of ETE1,2 from the construction area of ETE3,4.

2.1.2.2.9 Transport Infrastructure

Only one railway line enters into the territory of the Týn nad Vltavou Municipality with Extended Powers. The railway line from Temelín NPP runs through Temelín and Bohunice to Týn nad Vltavou. The above stated railway constitutes a part of the regional non-electrified single-track line number 192 Číčenice – Týn nad Vltavou. The section of line No. 192 runs through the vicinity of the site of the territory, i.e. within a distance of 3 km from the border of ETE3,4, according to Decree No. 215/1997 Coll. [L. 1].

The distance of the nearest railway lines from the Temelín NPP:

České Budějovice-Strakonice line section	10.3 km
České Budějovice - Veselí n.L. - Tábor	14.7 km
Tábor - Bechyně	12.6 km
Tábor - Písek - Ražice	18.9 km
Protivín - Čimelice line section	12.6 km
Číčenice - Týn n. Vlt. (from ETE3,4)	2.0 km
Číčenice - Prachatice	12.1 km
Dívčice - Netolice	10.4 km
Veselí n. Luž. - České Velenice	21.8 km

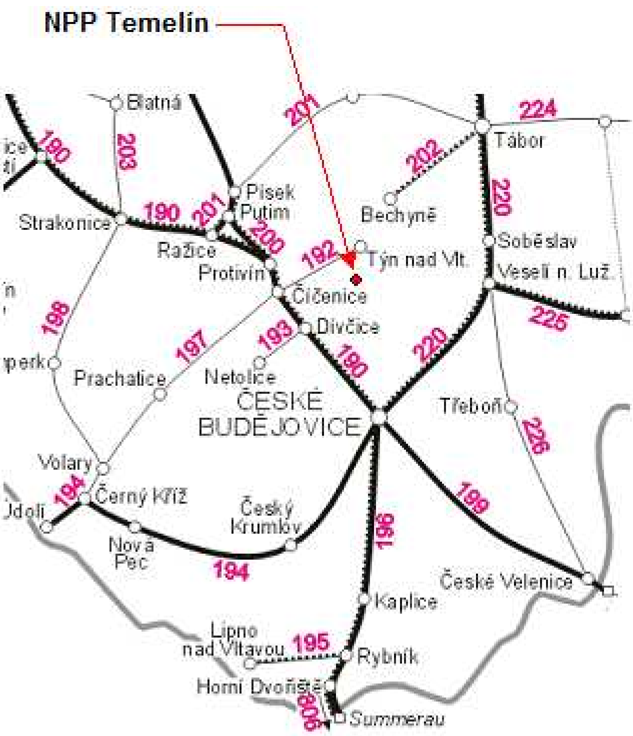

Fig. 4 Railway network at the Temelín site

Tab. 5 Protection zones of railways according to Act No. 266/1994 Coll.

Railway category	The protection zone of the railway consists of an area on both sides of the railway, with borders delimited by a vertical surface running
nation-wide railway for speed over 160 km/h,	100 m from the axis of the outer rail, but not less than 30 m from the border of the circumference of the railway,
nation-wide and regional railway	60 m from the axis of the outer rail, but not less than 30 m from the border of the circumference of the railway,
railway siding	30 m from the axis of the outer rail

The road transport infrastructure backbone, running through the affected area, is A-road No. II/105 in the segment between České Budějovice and Týn nad Vltavou. This road passes by the area of Temelín NPP in the southeastern direction, and the power plant is connected to it by a side road. A parking lot for cars of persons working in the power plant and of their visitors, including a bus station, is built in front of the power plant. Among other important roads there is A-road No. II/138, which connects to A-road No. II/105 south of the power plant and passes along its southwestern side towards the village of Temelín and continues to Písek to connect to A-road No. II/121 to Milevsko. The nearest road infrastructure also includes road No. II/141 in the Vodňany - Týn nad Vltavou segment, connected to A-road No. II/105 by an access road northeast of the power plant.

An internal road network has been built on the premises of Temelín NPP providing access to the various structures of the power plant.

The above mentioned roads No. II/105, II/138 and II/141 run through the zone of the vicinity of the site, i.e. within a distance of 3 km from the border of ETE3,4, according to Decree No. 215/1997 Coll. [L. 1]

The territory of the Týn nad Vltavou Municipality with Extended Powers is not directly connected to the existing network of motorways and highways. The nearest of them is the constructed part of the D3 motorway to the east of Tábor in the section of Chotoviny – Tábor South (about 35 km from the Temelín NPP). The construction of that motorway will continue towards the south to the border with Austria. No primary road runs through the territory of the Týn nad Vltavou Municipality with Extended Powers.

The next foundation of the road system in the vicinity of the Temelín NPP consists in international road E55 (I/3 Mirošovice - Dolní Dvořiště - Austria), the route of road E49 (I/20 Plzeň - České Budějovice) and the route of road I/29. The above stated road network runs in a zone of 10 – 20 km from the Temelín NPP.

Tab. 6 Protection zones of roads

Road category	Protection zone
motorway, highway, local highway	100 m from the axis of the adjacent lane
primary road and local primary road	50 m from the axis of the roadway or of the adjacent lane
secondary road, third class road and local secondary road	15 m from the axis of the roadway or of the adjacent lane

Note: According to Act No. 13/1997 Coll., the protection zone shall mean an area delimited by vertical surfaces up to a height of 50 m and at the distance indicated in the table above.

2.1.2.2.10 Technical Infrastructure

A high pressure gas pipeline runs through the territory of the Týn nad Vltavou Municipality with Extended Powers. The gas pipeline runs past the northwest edge of the surface intended for construction of ETE3,4. It is a corridor with 4 high pressure gas pipelines (DN 200, DN 800, DN 1000 and DN 1400 piping) containing natural gas of Russian origin.

Tab. 7 Protected ranges of gas pipeline

Gas pipeline	Average	Protected range	Security zone
High-pressure gas pipeline up to 40 bar	up to DN250	4 m	20 m
High-pressure gas pipeline above 40 bar	above DN500	4 m	160 m[7)]

Note to Tab. 7. Act No. 458/2000 Coll., [L. 217] establishes the two below listed zones for gas equipment:

- *Protected ranges shall serve to protect gas equipment and ensure its safe and reliable operation.*
- *Security zones are designed to prevent or alleviate the consequences of accidents occurring in the gas equipment, if any, and to protect the lives, health and property of persons.*

No product pipeline for transport of oil and fuels runs through the vicinity of the site of the territory intended for siting of ETE3,4. The nearest oil pipeline runs through the cadastral area of the municipalities of Bečice, Krakovčice, Hrušov, Žimutice, Štipoklasy, Modrá Hůrka, Bzí, Tuchonice and Radonice. Identification of the specific course of its route is not public information, due to safety aspects.

Overhead routes of power lines of extra high voltage of 400 kV consist of two routes running from ETE1,2 to the south to the switchyard in Kočín, reaching into the cadastral area of the municipalities of Chvalešovice and Kočín. Another route of extra high voltage of 400 kV runs to the east from the above stated power substation through the cadastral area of the municipalities of Kočín, Knín, Březí near Týn nad Vltavou, Litoradlice and Pořežany. ETE1,2 is further interconnected with the switchyard in Kočín with very high voltage lines of 2x110 kV. Another very high voltage line of 110 kV (about 1 km east of the Temelín NPP) runs to the northeast from the switchyard in Kočín towards the cadastral area of Nuzice and Litoradlice. The very high voltage routes run above the ground.

[7] Act No. 458/2000 Coll. [L. 217], in the wording valid at the time of issue of this report, determines the size of the safety zone of 160 m for high pressure gas pipelines and gas pipe connections with pressure above 40 bar and DN above 700. However, the gas pipelines were constructed at the time when the safety zone of 200 m applied to them. According to the temporary provisions under Section 98(3) of the valid wording of the act cited, "the safety zones for gas equipment determined according to the existing legal regulations and the previous written approvals of the construction of sites in such zones remain in force even after the day of this act coming into force". Therefore, the layout of location of the sources of external influences (see Annex Dwg. 4 to this Initial Safety Analysis Report) shows the depiction of the safety zone up to 200 m.

Tab. 8 Protected ranges of overhead line

Voltage	Protected range
above 35 kV to 110 kV including	15 m
above 220 kV to 400 kV including	25 m

Note: (1) In accordance with the provisions of Section 46(3) of Act No. 458/2000 Coll., [L. 217]: The protected range of the equipment of the grid is the area in the immediate vicinity of the equipment; the protected range is designed to secure reliable operation and protect lives, health and property of persons.

(2) The described lines were constructed before the entry of Act No. 458/2000 Coll. [L. 217], into force and are covered by the protected range with dimensions in accordance with Act No. 79/1957 Coll., on Production, Distribution and Consumption of Electricity (the Electricity Act) and its implementing Government Decree No 80/1957 Coll.

2.1.2.2.11 TV and Radio Transmitters

Such devices and their protection zones do not occur on lands for siting of ETE3,4 (see the statement of the Czech Telecommunications Office [L. 45]).

2.1.2.2.12 Protection Zones of Airports, Take-off and Landing Areas

Airports on the territory of the Czech Republic can be divided into several categories by traffic type (related to the equipment and instrumentation of the relevant airports). Generally speaking, the bigger airports have at the same time instrumentation of higher category and are able to meet standards of a higher category.

The IFR traffic shall mean such instrumentation of the airport (and in the airplane) that allows safe traffic – take-offs, landings and movements (rolling) of aircrafts even under conditions of reduced visual range.

Airports intended (equipped) for VFR (Visual Flight Rules) traffic do not have instrumentation and light equipment in complete form. It is expected that the pilots determine their position only by sight, without the help of devices. No instrumentation is required at the airport or in the aircraft. All further considered airports in the surroundings of the Temelín NPP are airports intended for VFR traffic. The runways at VFR airports can be made of concrete or asphalt (that is the case of airports that were formerly used by the army, such as the České Budějovice LKCS airport or some bigger air club airports) or of grass. Only the minimum bearing capacity of the runway is prescribed so that aircrafts with weight up to 5,700 kg can land on it. The runway length must meet the minimum requirements on traffic of common GA aircrafts, i.e. it usually exceeds 700 m. Some VFR airports may be equipped with runway lights (threshold, boundary and central lights, PAPI indication of descent level); then they are certified as airports for VFR NIGHT traffic. Only the Hosín airport – LKHS – is equipped for night traffic in the surroundings of Temelín.

The nearest airport to the Temelín NPP site is situated at a distance of about 13 km from the power plant. It is the defunct military airport of Bechyně. The next nearest airport is the Hosín airport at a distance of about 18 km from the Temelín NPP. The table below shows the airports in a zone of 10-40 km from the Temelín NPP.

Areas for the traffic of SLZ (air recreational vehicles) are approved by the Light Aircraft Association; they include areas for take-off and landing of ultralight aircraft. No responsible person is required; the demands on the equipment of the area are markedly lower; the limits of obstacles are milder. The airport does not need to be equipped with a radio station or with signal lights; a traffic log does not need to be kept. The areas may be registered (certified) by the Light Aircraft Association, which means that a Light Aircraft Association inspector has checked the area from the point of view of traffic safety; uncertified areas are operated without such inspection Both area types may have the status of a public or non-public (private) area. Arrival and landing in non-public areas is usually conditioned by the consent of the owner obtained in advance.

Tab. 9 Overview of airports in the surroundings of Temelín NPP

Name	ICAO code	Coordinates		Runway	Runway orientation	Runway surface
		Longitude	Latitude			
GA Airport						
České Budějovice	LKCS	N48 56 47	E014 25 39	2,500 m	27/09	Concrete
Hosín	LKHS	N49 02 20	E014 29 15	1,000 m/800 m	06/24	Grass/Asphalt
Soběslav	LKSO	N49 14 48	E014 42 49	740	18/36	Grass
Strakonice	LKST	N49 15 08	E013 53 45	900 m/780 m	03/21,13/31	Grass
Strunkovice	LKSR	N49 04 59	E014 04 33	900 m	15/33	Grass
Tábor	LKTA	N49 23 28	E014 42 30	1,100 m/850 m	12/30,16/34	Grass
Jindřichův Hradec	LKJH	N49 09 03	E014 58 18	700 m/760 m	07/25	Asphalt/Grass
Closed (former) military aerodromes						
Tábor Všechov	(LKTV-U-X)	N49 26 18	E014 37 17	2,000 m	13/31	Concrete
Bechyně	LKBC-X	N49 16 25	E014 30 06	2,400 m	12/30	Concrete
Blatná	LKBL-X	N46 25 51	E014 47 41	2,000 m	08/26	Concrete

Tab. 10 Overview of areas for air recreational vehicle traffic in the surroundings of the Temelín NPP

Areas for air recreational vehicle traffic, indicated in airports database						
Černovice	LKCERN			In 2010, closed for reclamation.		
Doudleby	LKDOUD	N48 53 06	E014 29 11	500 m	07/25	Grass
Frymburk	LKFRYM	N48 41 55	E014 11 51	465 m	06/24	Grass
Kaplice	LKKAPL	N48 43 07	E014 27 02	450 m	08/26	Grass
III - Košice u Tábora	LKKOSI	N49 19 05	E014 44 57	400 m	01/19	Grass
Kramolín	LKKRAM	N48 54 55	E014 43 40	455 m	09/27	Asphalt
Sand	LKPK-U	N49 20 30	E014 06 45	600 m	16/34	Asphalt
Třeboň Dvorce	LKTRED	N49 00 01	E014 43 19	600 m	08/26	Grass
Velešín	LKVELE-X	N48 49 36	E014 26 57	Area closed, cancelled.		

The surrounding area up to about 40 km from the Temelín NPP includes 7 GA airports with VFR DAY traffic and areas for air recreational vehicle traffic (a number of them are former areas for agrochemical activity). The GA airports included in the assessment of the Temelín NPP site with regard to the crash of an aircraft are described in detail in Section 2.2.2.4.

The position of airport areas situated in the vicinity of the Temelín NPP is depicted in the layout of the vicinity of the Temelín NPP, see Fig. 9.

The Temelín NPP site itself is situated inside the prohibited zone LK P 2 with vertical boundaries defined from the earth's surface to 1,500 metres above sea level and a horizontal boundary defined as a circle with its centre in the power plant and a radius of 1.1 nautical mile, that is, 2 km. No aircraft is allowed to enter that air space.

The proposed construction site of ETE3,4 is not situated in a protection zone of airports intended for civil and military traffic [L. 200] and [L. 201].

2.1.2.2.13 Protection Zone of the Power Plant

The protection zone of the Temelín NPP was declared in compliance with the building Act No. 50/1976 Coll., valid at that time, through the regulation of the former District National Committee of České Budějovice of 26 September 1985 [L. 243] with delimitation according to Annex to the Decision No. 25/1985 of the Czechoslovak Nuclear Energy Commission of 14 March 1985 [L. 237]. Both documents define a regime inside the protection zone, excluding, among other things, permanent settlement of the population and implementation of buildings not related to the operation of the power plant. The delimitation, definitions, description of borders and the evidence that the protection zone defined in that manner meets the currently valid legislation are described in Section 2.1.1.3 of the Preoperational Safety Report ETE1,2 [L. 18].

That delimitation of the protection zone is respected also in the current Land Use Plan of the municipality of Temelín and its purpose corresponds also to the current legal regulation of local planning because, within the meaning of Section 83 of Act No. 183/2006 Coll. [L. 256]; it protects the nuclear installation against adverse environmental impacts, protecting at the same time the surroundings of the nuclear power plant against its adverse effects.

The adverse effects of the nuclear power plant, against which the population must be protected, include particularly potential leaks of radioactive substances in case of a radiation accident. In accordance with Section 4 of Decree No. 215/1997 Coll. [L. 1], one of the exclusion criteria is the "unfeasibility of a timely introduction and complete implementation of all urgent measures to protect the population under the conditions of radiation accident of the installation or workplace, particularly with regard to the distribution of the population and to the presence of residential formations at the site corresponding to the expected siting". This criterion follows the protection of the population in the vicinity of the nuclear installation in case of accidental exposure situations where there would be danger of exceeding the guide values of intervention levels stated in Section 99(3) of Decree No. 307/2002 Coll. [L. 4]. Therefore, the extent of the protection zone corresponds to the need of creation of conditions for the opportunity to introduce timely and implement fully the urgent protective measures in the surroundings of the nuclear power plant. The concept of the approach to radiation protection for planned and accidental exposures is presented in more detail in Section 4.1.3 hereof; safety analyses that will be included in the following stages of the safety documentation shall check not only the adequacy of the technical solution with regard to the level of the risk related to the operation of the installation in normal, abnormal and accident conditions, but also the actual delimitation of the protection zone.

The protection zone is depicted in the map of the area in Annex, in Dwg. 1.

In the course of the construction and operation of the Temelín NPP, only the construction of the actual nuclear power plant, related buildings, site facilities, relocations of roads, railway lines and other utility lines caused by the construction of Temelín NPP can be implemented on the territory of the declared protection zone.

Additionally to the above stated protection zone declared by the Regulation of the former District National Committee of České Budějovice, ETE1,2 has a protection zone intended to provide reliable operation of the generating unit and protection of life, health and property of persons, resulting from Section 46(7) of Act No. 458/2000 Coll. [L. 217]. According to the cited provision of the Energy Act, the protection zone of the power generating unit is delimited by vertical planes passing at a horizontal distance of 20 m perpendicularly to the fencing or from the outer front of the external cladding of the power generating unit.

2.1.2.2.14 Basic Civic Amenities

The basic civic amenities in a 5 km zone around the Temelín NPP, i.e. in the on-site emergency planning zone, are concentrated in the town of Týn nad Vltavou. They include the town health centre, the city police and the fire brigade. School facilities found in a range of 5 km:

- Týn nad Vltavou - kindergarten, two higher elementary schools, grammar school
- Hněvkovice – secondary school, vocational school
- Neznašov – kindergarten, elementary school
- Dříteň – kindergarten and elementary school
- Temelín - kindergarten and lower elementary school

2.1.2.2.15 Use of Water in the Region

The main recipient of the region is the Vltava River, which is used to drain the whole territory. Small water courses serve only to drain the adjacent territories and they are used as outflow collectors of the drainage system to the minimum degree.

The population uses the Vltava River only for recreational purposes. For the future, it is considered to make the river navigable for recreational ships up to about 300 t from the lower reaches up to České Budějovice.

At a distance up to 20 km from the Temelín NPP, there are no recreational facilities or resorts significant from the point of view of migration of inhabitants.

The supply of drinking water to most inhabitants of the region is performed centrally. The crucial source of central supply consists in the group water mains from Římov. Římov is a water reservoir on the Malše River near Římov, which started to be used in 1977. The handling rules order to keep the minimum flow rate under the dam of the reservoir at 650 l/s. The flooded surface of the Římov Reservoir stretches into the cadastres of the municipalities of Římov, Svatý Jan nad Malší and Velešín. The reservoir serves as a drinking water source for České Budějovice and other towns and municipalities from the surroundings of the Temelín NPP; they are specifically listed in Tab. 11.

Another site providing central drinking water supply is Dolní Bukovsko (ground water taken with the help of wells).

An overview of water sources in the emergency planning zone and, in some cases, in its vicinity is shown in Tab. 11 [L. 48]. Four variants of water sources are distinguished:

- Bukovsko group water main
- Římov group water main
- "Municipal" water main to supply the municipality inhabitants who take water from local wells
- Own wells as an individual source of water supply for individual residences of citizens or seats of companies

Tab. 11 Drinking water supply in the emergency planning zone

Direction	City/municipality	District	Drinking water supply			
			Bukovsko	Římov	Water main for municipality	Own wells
NE	**Hodonice**	TA			X	approximately 1/3 of residents
NE	**Záhoří**	TA		X		
NE	**Březnice**	TA				X
NNE	*B*echyně	TA			Bechyně water main	
NNE	- Hvožďany	TA			Bechyně water main	
	- Senožaty	PE			Bechyně water main	
W, WNW	**Protivín**	PI		X		
W	- Krč	PI		X		
W	- Těšínov	PI		X		
W	- Záboří	PI		X		
W	- Milenovice	PI		X		
WNW	- Myšenec	PI		X		
WNW	**Žďár**	PI		X		
WNW	- Žďárské Chalupy	PI				X
WNW	- Nová Ves u Protivína	PI				X
NW	**Tálín**	PI		X		
NW	- Kukle	PI				X
NW	**Paseky**	PI				X
NW	- Nuzov	PI				X

Direction	City/municipality	District	Drinking water supply			
			Bukovsko	Římov	Water main for municipality	Own wells
NNW	**Albrechtice nad Vltav.**	PI		X		
NNW	- Újezd	PI		X		
NNW	- Hladná	PI		X		
NNW	- Údraž	PI				X
	- Jehnědno	PI				X
	- Chřesťovice	PI				X
W	***Vodňany***	ST		X		
W	- Čavyně	ST		X		
WSW	**Číčenice**	ST		X		
WSW	- Strpí	ST		X		
WSW	- Újezdec	ST		X		
N, NNW	***Všemyslice***	CB		X		
N	- Neznašov	CB		X		
NNW	- Všeteč	CB		X		
NNW	- Slavětice	CB		X		
N	- Bohunice (part of Všemyslice municipality)	CB		X		
E, ESE	***Dolní Bukovsko***	CB	X			
E	- Bzí	CB	X			
ESE	- Radonice	CB	X			
E	**Modrá Hůrka**	CB		X		
E	- Pořežánky	CB		X		
N	**Chrášťany**	CB				X
N	- Doubravka	CB			X	
N	- Koloměřice	CB				X
N	- Doubrava	CB				X
N	- Pašovice	CB				X
N	**Hosty**	CB				X
NE	Zvěrkovice (part of Temelín municipality)	CB		X		
ESE	Litoradlice (part of Temelín municipality)	CB		X		
SSW	Kočín (part of Temelín municipality)	CB		X		
WSW	Sedlec (part of Temelín municipality)	CB		X		

Direction	City/municipality	District	Drinking water supply			
			Bukovsko	Římov	Water main for municipality	Own wells
W	Lhota pod Horami (part of Temelín municipality)	CB		X		
NW	Temelín (municipality)	CB		X		
NE, E, ENE	***Žimutice***	CB	X			
NE	- Smilovice	CB	X			
ENE	- Třitim	CB	X			
ENE	- Hrušov	CB	X			
ENE	- Krakovčice	CB	X			
E	- Sobětice	CB	X			
E	- Tuchonice	CB	X			
E	- Pořežany	CB	X			
NE	**Čeňkov u Bechyně**	CB				X
ENE	**Dobšice**	CB		X		
E	**Horní Kněžeklady**	CB				X
E	- Štipoklasy	CB				X
ENE	**Bečice**	CB	X			
NNE, NE, ENE	***Týn n/Vlt.***	CB	X	X		
NNE	- Koloděje nad Lužnicí	CB	X	X		
NNE	- Vesce	CB	X	X		
NNE	- Netěchovice	CB	X	X		
NNE	-Nuzice	CB	X	X		
NE	-Předčice	CB	X			
	Hněvkovice (part of Týn nad Vltavou municipality)	CB	X			
ESE, SE, S	***Hluboká n/Vlt.***	CB		X		
ESE	- Hroznějovice	CB		X		
ESE	- Kostelec	CB		X		
SE	- Purkarec	CB	X			
SE	- Líšnice	CB				X
SE	- Jeznice	CB				X
SE	- Poněšice	CB				X
	- Bavorčovice	CB	X			
S	- Munice	CB		X		

Direction	City/municipality	District	Drinking water supply			
			Bukovsko	Římov	Water main for municipality	Own wells
SSE, S	***Olešník***	CB		X		
SSE	- Chlumec	CB		X		
S	-Nová Ves	CB				X
S	**Mydlovary**	CB		X		
S	**Zahájí**	CB		X		
SW, WSW	***Dříteň***	CB		X		
SW	- Strachovice	CB		X		
SW	- Záblatí	CB		X		
SW	- Záblatíčko	CB		X		
SW	- Radomilice	CB		X		
WSW	- Chvalešovice	CB		X		
SSW	Libiv (part of Dříteň municipality)	CB		X		
S	- Velice	CB		X		
WSW	Malešice (part of Dříteň municipality)	CB		X		
S	**Zliv**	CB		X		
SE	**Vlkov**	CB			X	
SSW	**Dívčice**	CB		X		
SSW	- Zbudov	CB		X		
SSW	- Dubenec	CB		X		
SSW	- Česká Lhota	CB		X		
SSW	- Novosedly	CB		X		
SSW	**Nákří**	CB		X		

The factors of water dilution in flow and retention of water in the reservoirs of the Vltava cascade are crucial for the potential impact of ETE3,4 on the population using water from the Vltava River. Another factor applied in spreading of radionuclides in a watercourse consists in their sorption in deposits and sediments. The process of spreading of contaminants in the Vltava Cascade and in the Elbe River down to the Hřensko/ Schőna profile is described in Section 5 of the report [L. 246]. The cited report shows that the impact of radionuclides in the water course including contamination from accident leaks from the Temelín NPP is negligible.

2.1.2.3 DEMOGRAPHIC DATA

2.1.2.3.1 Permanent Settlement

According to the result of the census published as at 31 December 2010 [L. 220], average density of population of the region is 63.52 inhabitant/km^2, which constitutes about 48% of the average of the whole republic. In the five-km emergency planning zone, the average number of inhabitants is only 30 inhabitants/km^2.

The development of the number of inhabitants in the South Bohemian Region in the last 50 years has been slightly increasing, see Tab. 12.

Tab. 12 Development of the number of inhabitants in the South Bohemian Region [L. 18][L. 220]

Year:	1961	1970	1980	1991	1996	2001	2003	2007	2010
Number of the population	573,711	577,543	613,171	622,890	627,580	625,267	624,889	633,264	638,706

The current development of age composition of the inhabitants in the South Bohemian Region corresponds to the nation-wide trend of an increasing share of older inhabitants in the total number of inhabitants, see Tab. 13.

Tab. 13 Development of age structure of inhabitants of the South Bohemian Region [L. 18]

Year		01/03/1961	01/12/1970	01/11/1980	03/03/1991	01/03/2001	26/03/2011
0–14 years	Men	-	63,982	74,098	68,843	53,305	47,691
	Women	-	60,490	70,879	65,549	50,526	45,596
	Total	146,153	124,472	144,977	134,392	103,831	93,287
15-64	Men	-	188,223	193,099	207,483	220,103	224,628
	Women	-	189,175	190,744	204,261	216,299	218,545
	Total	364,297	377,398	383,843	411,744	436,402	443,173
65 and over	Men	-	29,303	32,726	29,177	33,630	41,744
	Women	-	46,370	51,625	47,576	51,404	59,256
	Total	63,263	75,673	84,351	76,753	85,034	101,000
Total population		573,713	577,543	613,717	622,889	625,267	637,460

The distribution of population in the vicinity of the Temelín site was tabulated, in compliance with the Preoperational Safety Report [L. 18], within a radius of 100 km from the Temelín NPP.It shows the numbers of inhabitants in circular vicinity of the Temelín NPP site (the source for determination of the number of inhabitants is the statistical report of the Czech Statistical Office [L. 46] "Number of inhabitants in municipalities and towns of the Czech Republic as at 1 January 2010"). The number of inhabitants is, in compliance with para. 4.13 of the requirements of the IAEA standard NS-R-3 [L. 6], divided into 16 directions, each direction corresponding to a sector of 22.5° (direction 1 corresponds to north, further numbering runs in the direction of movement of clock hands). The sectors of the south and southwest directions at distances of about 70-100 km include the territories of Austria and Germany. The radius of about 95 km from the Temelín NPP includes bigger town agglomerations of Passau (Germany) – 50,537 inhabitants and Linz (Austria) – 189,500 inhabitants [L. 274].

Note: the calculation made use of the relative distribution of inhabitants of the individual municipalities into basic settlement units, derived from the status of the population as at 1 March 2001.[8]

[8] According to the communication of the Regional Analyses and Information Services Manager of the Czech Statistical Office, branch office in České Budějovice, the numbers of the population from the

Tab. 14 Number of the population by sectors and depending on distance from the Temelín NPP

Distance [km]		Number of the population in the direction from Temelín NPP																
From	To	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	**Total**
0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
1	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	197	0	197
2	3	0	0	0	0	0	0	0	0	**86**	0	0	0	0	0	197	0	283
3	4	0	180	0	94	0	0	8	0	0	0	104	71	165	0	0	0	622
4	5	147	0	0	0	0	57	0	0	0	795	0	0	0	0	0	0	999
5	6	0	549	6,750	0	82	0	37	0	59	0	78	157	0	0	0	0	7,712
6	7	18	54	82	88	0	0	0	0	0	141	0	71	0	0	0	108	562
7	8	36	13	0	0	35	0	152	9	136	0	216	0	0	90	0	0	687
8	9	24	295	54	112	28	170	144	0	0	593	1	135	0	0	0	0	1,556
9	10	10	48	103	207	251	0	33	0	29	19	153	94	157	221	0	461	1,786
10	11	60	0	0	46	88	28	0	0	160	553	189	220	0	0	11	18	1,373
11	12	118	117	116	99	93	0	4	29	0	0	161	26	314	1,848	114	0	3,039
12	14	508	164	248	163	305	161	286	0	1,696	2,445	246	0	208	2,322	403	114	9,269
14	16	134	1,745	3,566	104	34	1,098	362	1,431	2,539	345	333	955	5,934	42	62	102	18,786
16	18	383	262	461	185	321	446	1,407	831	242	282	307	468	1,167	269	133	395	7,559
18	20	349	319	235	31	569	476	267	2,856	1,273	538	1,843	346	271	238	677	272	10,560
20	22	1,323	213	88	649	0	214	109	3,427	19,184	887	1,156	238	219	579	16,191	924	45,401
22	24	627	600	493	375	5,541	74	58	5,506	41,324	835	125	193	1,433	358	13,464	503	71,509
24	26	508	1,001	1,436	3,692	1,490	328	3,235	11,525	17,474	930	96	935	298	1,127	264	112	44,451
26	28	206	1,542	282	3,982	1,839	2,147	169	3,346	6,957	569	1,563	1,080	318	615	666	406	25,687
28	30	7,745	375	1,383	2,203	671	172	975	2,761	5,076	446	435	446	588	1,245	1,152	374	26,047
30	35	2,692	1,672	42,436	4,824	3,166	1,021	7,525	3,632	3,733	4,333	961	15,367	1,441	6,280	1,335	892	**101,310**
35	40	2,112	3,169	3,451	2,426	951	1,238	1,982	4,417	6,596	3,875	376	1,460	5,167	21,845	3,263	2,164	**64,492**
40	45	2,622	2,674	2,210	2,978	2,457	3,344	1,107	6,341	3,523	13,366	93	812	4,542	3,157	2,377	2,171	**53,774**
45	50	2,689	3,684	2,417	3,351	2,901	21,509	5,898	1,751	7,177	5,166	168	4,192	8,858	2,832	7,521	2,181	**82,295**
50	55	7,253	10,542	2,905	1,878	8,301	1,558	1,306	4,983	3,453	1,245	2,475	1,402	3,493	7,893	3,521	6,669	**68,877**
55	60	3,638	4,046	1,947	7,235	5,510	7,553	394	292	2,844	1,173	107	741	1,040	4,200	3,491	7,475	**51,686**
60	65	4,299	6,611	3,599	3,432	4,566	3,067	9,123	154	1,213	5,047	254	810	2,705	14,073	1,877	41,107	**101,934**
65	70	15,105	5,579	14,942	19,395	4,900	3,959	9,932	623	2,975	5,423	11,441	1,604	1,409	4,217	5,825	4,214	**111,543**
70	75	5,378	24,509	5,229	4,368	4,125	5,809	3,279	586	9,843	7,659	8,914	9,234	214	3,415	5,818	3,277	**101,657**
75	80	15,083	13,112	5,025	4,747	11,109	13,601	3,294	4,523	8,714	2,245	23,507	13,689	14	5,070	11,494	13,842	**149,069**
80	85	23,189	20,152	9,290	13,906	12,268	5,854	1,489	1,705	10,439	2,160	9,104	15,690	2,025	25,764	20,441	22,260	**195,736**
85	90	52,084	29,187	7,808	8,607	15,237	11,799	13,409	1,753	8,250	3,660	27,529	6,248	359	7,577	22,421	13,227	**229,155**
90	95	126,971	93,647	7,732	14,200	48,207	4,440	1,100	2,415	37,862	6,330	15,350	13,408	5,383	8,832	78,682	8,163	**472,722**
95	100	345,143	202,950	6,766	27,998	10,388	4,387	1,429	1,611	56,793	8,326	50,627	14,260	7,875	5,992	115,170	2,851	**862,566**
0	**100**	**3 745**	**3 615**	**2 291**	**2 282**	**3 570**	**2 218**	**3 064**	**2 588**	**971**	**7 005**	**4 557**	**6 816**	**3 177**	**2 540**	**3 038**	**4 811**	**6 272**

The 5 km on-site emergency planning zone was chosen for the number of inhabitants in close vicinity of the Temelín NPP to provide for measures for preparation and accomplishment of evacuation of the population in case of a radiation accident (Decree No. 307/2002 Coll. [L. 4]). Tab. 15 includes the towns and municipalities and their parts falling into the on-site emergency planning zone and the numbers of inhabitants in the individual towns and municipalities. The numbers of inhabitants come from the data of the towns and municipalities from April 2011 [L. 238], [L. 239], [L. 240], [L. 241] and [L. 242]. From the total number of 9,813 inhabitants of the on-site emergency planning zone, 77.5% live in Týn nad Vltavou.

Tab. 15 Towns and municipalities and their parts situated in the on-site emergency planning zone (5 km from the Temelín NPP)

Name of the municipality or attached part of the municipality	**Number of the population**
Bohunice (part of Všemyslice municipality)	180
Týn nad Vltavou (municipality)	7,607
Zvěrkovice (part of Temelín municipality)	94
Hněvkovice (part of Týn nad Vltavou municipality)	82
Litoradlice (part of Temelín municipality)	57
Nová Ves (part of Olešník municipality)	41
Dříteň (municipality)	795
Libiv (part of Dříteň municipality)	18
Kočín (part of Temelín municipality)	86
Malešice (part of Dříteň municipality)	104
Sedlec (part of Temelín municipality)	71
Lhota pod Horami (part of Temelín municipality)	165
Temelín (municipality)	394
Všemyslice (municipality)	119

Tab. 16 indicates the numbers of inhabitants in individual sectors of the off-site emergency planning zone (the surface of the annulus from about 5 to 13 km from the NPP). The table consists of 16 parts corresponding to the individual sectors into which the zone is divided. The names of the towns and municipalities are in bold print; the names of villages (attached part of towns and municipalities) are in normal print. The names of towns and municipalities falling into 2 sectors or some parts (villages) of which do not fall into the emergency planning zone any more or are in the on-site zone are in bold italic print and the zero number of inhabitants is assigned to them.

Tab. 16 Numbers of the population in the off-site emergency planning zone

Sector: N **Name of municipality (village)**	**Number of the population**
Všemyslice	0
- Neznašov	535
Hosty	162
Chrášťany	390
- Doubravka	119
- Koloměřice	105
- Doubrava	64
- Pašovice	25
Number of the population in the sector	1 400

Sector: NE **Name of municipality (village)**	**Number of the population**
Týn n/Vlt.	0
- Předčice	66
Žimutice	0
- Smilovice	105
Hodonice	127
Březnice	207
Záhoří	63
Čeňkov u Bechyně	57
Number of the population in the sector	625

Sector: ENE **Name of municipality (village)**	**Number of the population**
Bečice	104
Dobšice	116
Žimutice	215
- Třitim	20
- Hrušov	20
- Krakovčice	40
Number of the population in the sector	515

Sector: E **Name of municipality (village)**	**Number of the population**
Žimutice	0
- Sobětice	40
- Tuchonice	30
- Pořežany	110
Horní Kněžeklady	88
- Štipoklasy	52
Modrá Hůrka	51
- Pořežánky	39
Dolní Bukovsko	0
- Bzí	54
Number of the population in the sector	464

Sector: N **Name of municipality (village)**	**Number of the population**
Hluboká n/Vlt.	0
- Hroznějovice	53
- Kostelec	153
Dolní Bukovsko	0
- Radonice	79
Number of the population in the sector	285

Sector: SE **Name of municipality (village)**	**Number of the population**
Hluboká n/Vlt.	0
- Purkarec	152
- Líšnice	26
- Jeznice	35
- Poněšice	36
Vlkov	20
Number of the population in the sector	269

Sector: SSE Name of municipality (village)	Number of the population
Olešník	0
- Chlumec	125
Number of the population in the sector	125

Sector: S Name of municipality (village)	Number of the population
Olešník	559
Mydlovary	267
Zahájí	375
Dříteň	0
- Velice	133
Hluboká n/Vlt.	0
- Munice	204
Zliv	3,600
Number of the population in the sector	1 542

Sector: SSW Name of municipality (village)	Number of the population
Dívčice	255
- Zbudov	98
- Dubenec	88
- Česká Lhota	50
- Novosedly	48
Nákří	212
Number of the population in the sector	751

Sector: SW Name of municipality (village)	Number of the population
Dříteň	0
- Strachovice	65
- Záblatí	115
- Záblatíčko	75
- Radomilice	65
Number of the population in the sector	320

Sector: WSW Name of municipality (village)	Number of the population
Dříteň	
- Chvalešovice	150
Číčenice	369
- Strpí	35
- Újezdec	44
Number of the population in the sector	598

Sector: W Name of municipality (village)	Number of the population
Protivín	0
-Krč	217
- Těšínov	86
- Zábořı́	152
- Milenovice	165
Vodňany	0
- Čavyně	27
Number of the population in the sector	647

Sector: WNW Name of municipality (village)	Number of the population
Protivín	4,013
- Myšenec	209
Žďár	172

- Žďárské Chalupy	53
- Nová Ves u Protivína	10
Number of the population in the sector	448

Sector: NW **Name of municipality (village)**	**Number of the population**
Tálín	135
-Kukle	24
Paseky	135
- Nuzov	7
Number of the population in the sector	301

Sector: NNW **Name of municipality (village)**	**Number of the population**
Všemyslice	0
- Všeteč	100
- Slavětice	32
Albrechtice nad Vltav.	430
- Újezd	28
- Hladná	28
- Údraž	120
Number of the population in the sector	738

The total number of the population in the off-site emergency planning zone is 17,197.

2.1.2.3.2 Temporary Population

The region in the surroundings of the Temelín NPP within 30 km has varied landscape and offers very good conditions for tourism and recreation. According to [L. 18], there are about 6,000 places available on camping sites, chalet sites and hostels of the region. A capacity of a further 2,000 – 2,200 places is considered in public campsites along the Lužnice, Otava and Blanice rivers and at some ponds. Hotels of České Budějovice, Písek, Hluboká nad Vltavou and Vodňany have about 2,500 beds available. Additionally to the above stated accommodation opportunities in public facilities, there are about 5,500 buildings for individual recreation (chalets, lodges, etc.) within a radius of about 30 km from the Temelín NPP.

Provided that in the peak times of the main recreation season, i.e. in July and August, the above stated accommodation capacities are 70-80% utilized and that owners from outside the 30 km zone from the Temelín NPP come to a half of the buildings of individual recreation and 1 building is inhabited by 4 persons, it can be expected that the number of inhabitants in the main season will increase by about 11,000. Camping sites, chalet sites and public campsites will have the number of temporarily accommodated persons increased by about 6,500, and hotels by about 960 persons. Thus the total number of persons will increase by about 18,460.

The proposed principles of development of the South Bohemian Region [L. 218] do not include stimuli for quantitative changes of the above stated factors of temporary accommodation. The number of accommodation facilities, recreational objects, camping sites and chalet sites will remain at the current level; an increase of their quality level can be considered. The peak numbers in the season of recreation and holidays in 2025 – 2035 will oscillate between 18,000 – 20,000 above the number of 278.7 thousand permanently living inhabitants within a radius of 30 km from the Temelín NPP (see [

Tab. 16].

In view of the fact that the Temelín site does not have production or office buildings that are significant from the point of view of the number of employees, not related to the operation of the power plant, and that it can be expected that schools, with regard to their type, will be used mainly by local inhabitants, the capacity numbers of those facilities are not included into the balance of temporary settlement at the site. An exception from the above stated description consists in the High Professional School and Vocational School of Hněvkovice where there can be up to about 800 persons (approximately half of them commuting) and the operator and supplier companies at the Temelín NPP site with about 3,200 persons.

2.1.2.4 EXPECTED DEVELOPMENT OF THE REGION

2.1.2.4.1 Development of the Region According to the Land Use Plan

At the time of elaboration of this Initial Safety Analysis Report, the Regional South-Bohemia Authority as the recipient was finishing the Development Principles of the South Bohemian Region that constitute, from an administration perspective, the initial basis for implementation of all significant changes in the development and use of the area in question. The section covering the emergency planning zone of the Development Principles of the South Bohemian Region includes the plans summarized in Tab. 17.

Tab. 17 Supralocal and developmental areas in the Temelín region

KP15	**Týn nad Vltavou**, it is a commercial-industrial area delimited at the southern edge of Týn nad Vltavou next to the existing areas of non-residential character, as part of the main settlement centre at the southern edge of the specific area of supralocal significance, N-SOB1 Orlicko. Affected cadastral areas: Týn nad Vltavou.
KP16	**Ekopark Býšov**, not far from the Temelín NPP, within reach of all networks of technical infrastructure, a commercial-industrial developmental area of supralocal significance is proposed, with prevailing contents of production and use of renewable energy sources in the form of a bio-ethanol plant, combined with photovoltaic power plant, biogas station and turbine running on bio-fuels. Affected cadastral areas: Knín.
KP38	**Temelín** – areas for completion of Temelín NPP Units 3 and 4, in compliance with the Territorial Development Policy of the Czech Republic 2008, proposed in relation to the existing premises of Temelín NPP. Affected cadastral areas: Temelín, Temelínec, Křtěnov, Březí u Týna nad Vltavou.
SO7	**Týn nad Vltavou - north**, at the northern edge of Týn nad Vltavou, constituting the main centre of tourism and settlement at the southern edge of the specific area of supralocal significance, N-SOB1 Orlicko, an area of supralocal significance for mixed residential function is delimited. Affected cadastral areas: Týn nad Vltavou.
A 1	Reclamation area of Mydlovary (Mydlovary, Nákří, Olešník, Dívčice) in the area of waste repository from uranium ore processing DIAMO-MAPE (sludge lagoons). Affected cadastral areas: Olešník, Nákří, Dívčice, Mydlovary u Dívčic.
D18	Vltava Waterway – the plan to make the Vltava River navigable for ships up to 300 t, 45 m long and 6 m wide (lock chambers), from Jirásek Bridge of České Budějovice to the border with the Central Bohemian Region; the corridor is discontinuously delimited

	into 2 self-standing sections where adaptations of the river bed and vicinity are needed, consisting in digging the river bed to a depth of 160 cm (130 cm ship draught + 30 cm reserve) and constructions related to making the river navigable: D18/1, the section of České Budějovice – Hluboká nad Vltavou, starts by the turning basin between the Jirásek Weir and the Long Bridge, continuing with the end dock (tourist port), Lanna Shipyard under the Long Bridge (on the left bank), the tourist port of New Bridge (on the left bank), the protective port of České Vrbné (on the left bank), the new lock chamber of České Vrbné (on the right bank) and modernization of the weir of České Vrbné, further the freight port of Hrdějovice (on the right bank), related to the plan of the public logistic centre of České Budějovice - Nemanice, the sports port of Hluboká nad Vltavou (left bank above the weir of Hluboká nad Vltavou), the lock chamber at the weir in Hluboké nad Vltavou, corridor width of 100 m. D18/2, the section of Hněvkovice nad Vltavou – Týn nad Vltavou, the section starts with equipment of the lock chamber in the dam of the Hněvkovice Reservoir, modernization of the weir and construction of a lock chamber on the Hněvkovice Weir, digging of the bottom of the Kořensko dam and relocation of the historical bridge of Týn nad Vltavou upriver and construction of a new bridge (with underpass height of 5.25 m) on the place of the relocated bridge, corridor width of 100 m. Affected cadastral areas: České Budějovice 3, České Budějovice 2, České Vrbné, Hrdějovice, Bavorovice, Hluboká nad Vltavou, Litoradlice, Třitim, Hněvkovice u Týna nad Vltavou, Týn nad Vltavou, Všemyslice, Hosty, Pašovice.
D 20	České Budějovice Airport, area for equipment, activities and processes related to operation of an airport of state-wide importance with international operation, České Budějovice, on the territory of the municipality of Planá u Českých Budějovice, delimitation of territory for premises and areas related to operation of the airport. Additionally to the provision of conditions for domestic and international air traffic, conditions for related commercial, logistic, storage and other transport functions will be provided, including provision and organisation of cultural, sporting and social events.Affected cadastral areas: Homole, Planá u Českých Budějovic.
D 21	Strakonice Airport, area for equipment, activities and processes related to operation of a public domestic airport of supralocal significance. Affected cadastral areas: Nové Strakonice, Mutěnice u Strakonic.
D 22	Písek Airport, area for equipment, activities and processes related to operation of a public domestic airport of supralocal significance. Affected cadastral areas: Krašovice u Čížové.
D29	Road II/137 – the plan is closely related to the need for implementation of the capacity of the connection road of the construction site of Temelín NPP Units 3 and 4, and of motorway D3 near Tábor; the corridor for new sections, relocations and improvement of parameters of road II/137 including construction of a new road stretch is being delimited.
D31	Road II/141 – plan of new sections, relocations and homogenisation of the existing road, discontinuously delimited from Temelín (southern bypass) through Vodňany (crossing with road I/20) to Volary and it is subdivided into 6 self-standing sections: D31/1, southern bypass of Temelín, plan associated with completion of the Temelín NPP, to the south from built-up territory of the village, corridor width of 100 m.
Ee31	Very high voltage 110 kV Kočín – Veselí nad Lužnicí, plan of a new very high voltage line, corridor width of 100 m. Affected cadastral areas: Borkovice, Dolní Bukovsko, Horní Bukovsko, Bzí u Dolního Bukovska, Tuchonice, Jaroslavice u Kostelce, Litoradlice, Březí u Týna nad Vltavou, Knín, Kočín, Chvalešovice, Modrá Hůrka, Pořežany, Řípec, Sviny, Žíšov u Veselí nad Lužnicí.

Ee32	Extra high voltage 400 kV Temelín NPP – Kočín, plan of power outlet from Temelín NPP Units 3 and 4 to the switchyard in Kočín, a short section of extra high voltage line between the nuclear power plant itself and the switchyard; the plan is delimited by a corridor of funnel shape with varied width from 200 – 650 m. Affected cadastral areas: Chvalešovice, Temelinec, Kočín, Březí u Týna nad Vltavou.
Ee33	Extra high voltage 400 kV Kočín – Mírovka – plan of extra high voltage line 400 kV from the switchyard in Kočín to the switchyard in Mírovka, delimited on the territory of South Bohemian Region by a corridor of width of 200 m. Affected cadastral areas: Dříteň, Chvalešovice, Hartmanice u Žimutic, Jaroslavice u Kostelce, Štipoklasy, Modrá Hůrka, Březí u Týna nad Vltavou, Knín, Kočín, Litoradlice, Pořežany, Sobětice u Žimutic, Tuchonice, Žimutice, Nové Dvory u Pořína, Debrník, Hodětín, Choustník, Kajetín, Chrbonín, Klenovice u Soběslavi, Komárov u Soběslavi, Košice u Soběslavi, Krátošice, Krtov, Myslkovice, Kozmice u Chýnova, Sedlečko u Soběslavi, Rybova Lhota, Chabrovice, Skopytce, Nedvědice u Soběslavi, Brandlín u Tučap, Svinky, Vlastiboř u Soběslavi, Klečaty.
Ee35	Enlargement of the existing switchyard in Kočín, plan of enlargement of the area of the existing switchyard on the eastern and north-western side, including an area for watercourse relocation on the western edge of the delimited surface and draining measures on the northern side. Affected cadastral areas: Kočín, Chvalešovice
Et1	Long-distance heat piping of Temelín NPP – Olešník – Zliv – České Budějovice, plan of long-distance heat piping to supply the city of České Budějovice, the corridor is delimited from the premises of the nuclear power plant, east of Kočín, to the eastern border of the built-up area of Velice, along the western edge of the built-up area of Olešník, between Mydlovary and Zahájí, on the eastern edge of Zliv and further along the railway to the northern edge of the built-up area of České Budějovice where it is delimited up to the crossing of the Strakonická and Na Dlouhé louce Streets on the left bank of the Vltava River. Delimited by a corridor of a width of 100 m. Affected cadastral areas: Březí u Týna nad Vltavou, Knín, Kočín, Olešník, Hluboká nad Vltavou, Munice, Bavorovice, České Vrbné, České Budějovice 2.
PT 4	Albrechtice nad Vltavou – territorial reserve for future area of extraction of building stones, within the balanced exclusive deposit of Albrechtice. Affected cadastral areas: Albrechtice nad Vltavou.
URB	Landscape of a strongly urban environment – Occurrence: territory of town agglomerations including subdural conveniences of the towns of České Budějovice, Týn nad Vltavou, Jindřichův Hradec, Písek, Protivín, Strakonice, Vodňany, Soběslav, Tábor, Veselí nad Lužnicí and the area of the Temelín NPP, – Landscape character: town environment of historical centres, environment of industrial, residential and recreational zones with parks, buildings and line green vegetation of non-original species – Natural values: town and castle parks, accompanying green vegetation of watercourses, town green vegetation – Cultural values: historical structure of urban housing development, monumental buildings (included in city conservation area) – Aesthetic values: park areas with well-maintained green vegetation, historical housing development, monumental buildings.

The implementation of the plans described in the Development Principles of the South Bohemian Region is usually expected within a time horizon of 4 years. The position of the plans is marked in the enclosed layout of the area, see Dwg. 5.

2.1.2.4.2 Opportunities for Industrial Growth

The protection zone of the Temelín NPP, declared in 1985 by the former District National Committee, restricts the industrial growth in the immediate vicinity of the Temelín NPP and thus also the creation of further stationary sources of events capable of jeopardizing the power plant.

However, beyond the border of the protection zone, it can be expected that new business opportunities will be applied, creating potential new sources of events and risks.

With regard to the low population density, the country character and the remoteness of the vicinity of the Temelín NPP near region, it can be, however, expected that future industrial growth in the region will be gradual, dictated rather by regional interests and investments than by state-wide or even international interests. Such character of growth should be more favourable in terms of the determination of design parameters, because it can expectedly lead to origination of rather small sources of events, jeopardizing the operation of the Temelín NPP less.

In connection with the preconditions given by the Development Principles of the South Bohemian Region, the following trends with potential impact on the current list of external sources of events can be predicted:

- Continuing effort to revive, use and reconstruct abandoned or waste buildings and sites that served for industrial or agricultural production or to the army in the past. The construction at the Býšov site corresponds to that trend. Attention should be paid to further use of the area of the ZACHEMO company in Temelín and to the premises of the former military warehouse in Podhájí. Further opportunities are marked as commercial-industrial plans (PK) and mining industry (ET) in the Development Principles of the South Bohemian Region
- If the planned construction of the heat pipeline of Temelín NPP - České Budějovice is implemented (see also pos. Et1 in the Development Principles of the South Bohemian Region), it can expectedly also act as an impulse for origination of new businesses near the heat pipeline route. The availability of a big source of low-potential heat can attract some branches of modern business, e.g. biotechnological plants
- Development of transport infrastructure in parts that can increase the traffic load near the Temelín NPP (e.g. road II/141 stated in pos. D31 in the Development Principles of the South Bohemian Region)

Another group includes power investments related to the connection of ETE3,4 to the electrification system and to takeout of heat to České Budějovice (see pos. Ee and Et1 in Tab. 17). The above stated positions will be handled without having impact on the Temelín NPP.

The actual completion of the Temelín NPP Units 3 and 4 is included in the Development Principles of the South Bohemian Region within the commercial-industrial areas of supralocal significance KP38 (in compliance with the Territorial Development Policy of the Czech Republic, 2008).

The projects listed in the Development Principles within a radius of 15 km from the construction site of the Temelín NPP are shown in the enclosed layout Dwg. 5.

In further preparation of the Temelín NPP, the development in the vicinity of the power plan shall be continuously monitored and new risk sources that can potentially emerge in the future shall be identified and analysed at regular intervals.

2.1.2.4.3 Demographical Development

The conditions for the demographical development of the South Bohemian Region result from the above stated Development Principles [L. 218] and of the socio-demographic profile of the South Bohemian Region, the preparation of which has been started already, but no outputs were available at the time of completion of this report.

The Development Principles of the South Bohemian Region do not include projects with significant impact on permanent settlement in the vicinity of the Temelín NPP. New sporting and recreational facilities listed in the Development Principles of the South Bohemian Region do not have the character of mass facilities (e.g. the golf course of Týn nad Vltavou) that would significantly increase the number of temporary inhabitants in the vicinity of the Temelín NPP.

Also the existing timeline of the number of inhabitants in the South Bohemian Region and Týn nad Vltavou as the biggest town in the vicinity of the Temelín NPP confirms the stagnation of the number of inhabitants in the vicinity of the Temelín NPP.

For the above stated reasons, the demographical development was derived from the prognosis of the number of inhabitants of the South Bohemian Region, published by the Czech Statistical Office [L. 46]. The expected development of the number of inhabitants, published by the Czech Statistical Office, corresponds to a decline by 17.2% (see Tab. 18) within 50 years. The plans stated in the land use plan do not create stimuli for migration with profit for the South Bohemian Region that would transform the natural decrease into a significant increase of the number of inhabitants.

Tab. 18 Projection of inhabitants without impact of migration

Population movement	2015	2040	2065
Live-born	6,427	5,236	4,067
Deceased	6,436	7,648	7,815
Natural increase	-9	-2,412	-3,748
Live-born per 1,000 residents	10.1	8.7	7.7
Deceased per 1,000 residents	10.1	12.7	14.8
Total fertility	1.54	1.65	1.68
Estimated life expectancy at birth			
Men	76.0	81.7	86.7
Women	81.6	86.8	91.0
Number of the population (as at 01/01)	**638,419**	**602,277**	**528,500**

The change of age structure of the population results from the indicators of the projection of the number of inhabitants stated in Tab. 18. The decrease of the share of children and persons in productive age will be substituted by 63.4% growth of inhabitants over 65 years of age (Tab. 19) within 50 years.

Tab. 19 Development of age structure of the population

	2015	2040	2065
Total men of this age:	**315,386**	**296,279**	**260,478**
0 - 14	7.7%	6.4%	6.3%
15 - 64	34.1%	30.0%	26.9%
65 and over	7.6%	12.8%	16.1%
Total women of this age:	**323,033**	**305,998**	**268,022**
0 - 14	7.3%	6.0%	6.0%
15 - 64	33.0%	28.8%	25.6%
65 and over	10.2%	15.9%	19.2%

Decrease of total number of inhabitants has also occurred in the on-site emergency planning zone..

Tab. 20 Development in the number of the population in the on-site emergency planning zone

Name of the municipality or attached part of the municipality	Number of the population		
	March 2004	April 2011	Index 2011/2004
Bohunice (part of Všemyslice municipality)	185	180	97.3%
Týn nad Vltavou (municipality)	8,292	7,607	91.7%
Zvěrkovice (part of Temelín municipality)	89	94	105.6%
Hněvkovice (part of Týn nad Vltavou municipality)	81	82	101.2%
Litoradlice (part of Temelín municipality)	47	57	121.3%
Nová Ves (part of Olešník municipality)	38	41	107.9%
Dříteň (municipality)	650	795	122.3%
Libiv (part of Dříteň municipality)	15	18	120.0%
Kočín (part of Temelín municipality)	80	86	107.5%
Malešice (part of Dříteň municipality)	100	104	104.0%
Sedlec (part of Temelín municipality)	61	71	116.4%
Lhota pod Horami (part of Temelín municipality)	139	165	118.7%
Temelín (municipality)	359	394	109.7%
Všemyslice (municipality)	90	119	132.2%
Number of the population in the on-site emergency planning zone	10,226	9,813	96.0%

The described demographical development shows that change of number and composition of inhabitants does not constitute a limiting factor for ETE3,4 and that the demographical development does not bring stimuli for essential changes in emergency planning (see below Section 2.8 hereof).

2.1.3 REQUIREMENTS AND CRITERIA

Within the framework of geography and demography, the items specified below in Tab. 21 are assessed from the criteria set forth in Decree No. 215/1997 Coll. [L. 1], and the requirements stipulated in the IAEA standard NS-R-3 [L. 6].

Tab. 21 Criteria for site assessment in terms of geographical and demographical characteristics

ID	§	Requirement of the criterion in accordance with Decree No. 215/1997 Coll.	Para.	Requirements in the IAEA standard NS-R-3[9]
1.1	§ 4q)[10]	The lands selected for siting encroaching on protected zones of motorways and railways.		
1.2	§ 5l)[11]	Interference of routes and protection zones of gas, oil and product pipelines and underground stockpiles of raw materials transported to the plots selected for siting,		
1.3	§ 5m)	Occurrence of radio and television transmitters and their protection zones on lands for siting,		
1.4	§ 5n)	Protection zones of airports, 10) especially their take-off and landing areas and buildings with ground-based aerial equipment, encroaching on the vicinity zones of the site,		
1.5			4.10.	The distribution of the population within the region shall be determined.

[9] The requirements stipulated in the IAEA standard NS-R-3 [L. 6] either complement the substance of the criteria selected from the regulation of the State Office for Nuclear Safety for siting of nuclear installations (e.g. specify the method for applying the criteria) or represent an independent area for site assessment and formulation of design basis for the installation to be sited.

[10] Exclusion criteria definitely make impossible the utilisation of the area for siting.

[11] The conditional criteria enable to utilise the area or the land for siting provided that there is possible or available the technical solution of the unfavourable territorial conditions.

ID	§	Requirement of the criterion in accordance with Decree No. 215/1997 Coll.	Para.	Requirements in the IAEA standard NS-R-3[9]
			4.11	In particular, information on existing and projected population distributions in the region, including resident populations and to the extent of possible transient populations, shall be collected. This data should be maintained and kept up to date over the lifetime of the nuclear installation. The radius within which this data is to be collected should be chosen in accordance with the national practices in the state in question, with account taken of special situations. Special attention shall be paid to the population living in the immediate vicinity of the nuclear installation, to densely populated areas and population centres in the region, and to residential institutions such as schools, hospitals and prisons.
			4.12.	The most recent census data for the region, or information obtained by extrapolation of the most recent census data, shall be used in obtaining the population distribution. In the absence of such data, a special study shall be carried out.
			4.13	The mentioned data shall be analysed to give the population distribution in terms of direction and distance from the nuclear installation. An evaluation shall be performed of the potential radiological impacts of normal discharges and accidental releases of radioactive material, including reasonable consideration of released radioactive material due to severe accidents. To this purpose, site specific parameters shall be used as appropriate.

ID	§	Requirement of the criterion in accordance with Decree No. 215/1997 Coll.	Para.	Requirements in the IAEA standard NS-R-3[9]
1.6			4.14	The uses of land and water shall be characterized in order to assess the potential effects of the nuclear installation in the region and particularly for the purposes of preparing emergency plans. The investigation should cover land and water bodies that may be used by the population or may serve as a habitat for organisms in the food chain.

2.1.4 DOCUMENTS PROVIDING A BASIS FOR THE ASSESSMENT

The following documents were used to evaluate basic data of the near region (geography and demography):

- Preoperational Safety Report for Temelín NPP Units 1 and 2, [L. 18]
- Preparation of Land-Use Analytical Data for Administration District of Municipality with Extended Authorities Týn nad Vltavou, 12/2008 [L. 247]
- New Nuclear Installation at Temelín Site Including Power Outlet to the Switchyard in Kočín, Environmental Impact Assessment Documentation, SCES – Group, s.r.o., 05/2010 [L. 29]
- Landscape Character Assessment Study under Section 12 of Act No. 114/1992 Coll., on Protection of Nature and Landscape, on the Territory of the South Bohemian Region, Artech spol. s.r.o., G.L.I., 2009 [L. 262]
- Mgr. Petr Sojka: Update of air traffic data in the area surrounding the Temelin nuclear power plant - 2010, 12/2010 [L. 34]
- Assessment of the influence of maximal project accident of the long-distance gas pipeline on NNI buildings, technical report, UJV Řež, a.s. - ENERGOPROJEKT Division Prague, 10/2007 [L. 44]
- Website of the Czech Statistical Office http://www.czso.cz/csu/redakce.nsf/i/obyvatelstvo_lide [L. 220]
- Dříteň Municipality. Letter on the Number of the Population in Municipality Parts of 27 April 2011, addressed to ČEZ, a. s. [L. 238]
- Všemyslice Municipality. Letter on the Number of the Population in Municipality Parts of 29 April 2011, addressed to ČEZ, a. s. [L. 239]
- Olešník Municipality. Letter on the Number of the Population in Municipality Parts of 28 April 2011, addressed to ČEZ, a. s. [L. 240]
- Temelín Municipality. Letter on the Number of the Population in Municipality Parts of 20 May 2011, addressed to ČEZ, a. s. [L. 241]

- Týn nad Vltavou Municipality. Letter on the Number of the Population in Municipality Parts of 2 May 2011, addressed to ČEZ, a. s. [L. 242]

2.1.5 METHODS APPLIED TO THE EVALUATION

The method of special compilation of near-region related facts on the basis of reports and analyses created for the purposes of the Initial Safety Analysis Report together with the additional use of information from public information servers was applied to the evaluation of basic data of site (geography and demography). The applied method used the approaches specified in the requirements in para. 4.10 to 4.14 of the IAEA standard NS-R-3 [L. 6].

The assessment of transient populations at the Temelín near region defined in para. 4.11 if the IAEA requirements [L. 6] was based on the expected use of temporary accommodation potential in the area in question.

2.1.6 DEFINITION OF THE AREA EXAMINED

The area to be examined in the description about basic data of site (geography and demography) was delineated by the boundary of the South Bohemian Region, boundary of the České Budějovice District and boundary of the administrative district of the Týn nad Vltavou Municipality with Extended Powers.

Industrial production, energy sources, road, railway and water transportation and storage of hazardous materials were assessed on the area within 10 km from the boundary of the land proposed for siting of ETE3,4. In the case of the sources of risks affecting only in the form of fire, the radius of the examined area is 3 km. An area in the circuit of 40 km around ETE3,4 was delineated for mapping of airports. The mentioned approach complies with the IAEA recommendation in NS-G-3.1 [L. 9]. The delineation of the examined area for external influences is the subject of Section 2.2.6 hereof.

A five-km zone designed for emergency planning was selected for the assessment of the number of the population and basic civic amenities in the vicinity of Temelín NPP (see Decree No. 423/2001 Coll. [L. 3]) and an area with a radius of 100 km for statistical demographical overview in order to ensure relation to the Preoperational Safety Report [L. 18].

2.1.7 DETAILED ASSESSMENT OF ALL REQUIREMENTS AND CRITERIA DEFINED IN DECREE NO. 215/1997 COLL., IN COMBINATION WITH IAEA STANDARD NS-R-3

2.1.7.1 CRITERION UNDER SECTION 4(Q) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Section 4(q) of Decree No. 215/1997 Coll. [L. 1], is reproduced in Tab. 21 under item 1.1.

The protection zone of motorway, highway and local road is 100 m (see Tab. 6 hereof). No such road is located in the site vicinity up to 3 km from ETE3,4.

The railway protection zone is 100 m and 60 m from the axis of the outer railway track for national and regional lines, respectively. The nearest railway line is located 2 km from ETE3,4[12].

The siting of ETE3,4 is not in conflict with the exclusion criterion under Section 4(q) of Decree No. 215/1997 Coll. [L. 1]

2.1.7.2 CRITERION UNDER SECTION 5(L) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Section 5(l) of Decree No. 215/1997 Coll. [L. 1], is reproduced in Tab. 21 under item 1.2.

The protection zone of high-pressure gas pipelines is 4 m (see Tab. 7 hereof). The nearest gas pipeline is located 150 m from the northwest corner of the power plant (see Dwg. 4 hereof). The investigation performed shows that no oil pipelines, product pipelines and the underground stockpiles of transported raw materials are located in the vicinity of the site (up to 3 km).

The area of ETE3,4 does not encroach on the protection zones of gas, oil, product pipelines and the underground stockpiles of transported raw materials.

In terms of the requirements of criteria defined in Section 5(l) of Decree No. 215/1997 Coll. [L. 1], the siting of the power plant is not conditioned by measures in power plant design.

The layout of the structures in the area of ETE3,4 shall take account of safety risk of a long-distance gas pipeline as well as natural gas distribution line, assessed under the study [L. 44]. The risks are described in Section 2.2.2.5 and measures are specified in Section 2.10.

2.1.7.3 CRITERION UNDER SECTION 5(M) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Section 5(m) of Decree No. 215/1997 Coll. [L. 1], is reproduced in Tab. 21 under item 1.3.

There are no radio and television transmitters and their protection zones on lands for siting of ETE3,4 (see the statement of the Czech Telecommunications Office [L. 45]).

In terms of the requirements of criteria defined in Section 5(m) of Decree No. 215/1997 Coll. [L. 1], the siting of the power plant is not conditioned by measures in power plant design.

2.1.7.4 CRITERION UNDER SECTION 5(N) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Section 5(n) of Decree No. 215/1997 Coll. [L. 1], is reproduced in Tab. 21 under item 1.4.

The siting of ETE3,4 is not in conflict with the protection zone of the civil airport with the take-off and landing runway and clearway with length of 1,800 m and the protection zone of the take-off and approach areas of the military aerodrome in Bechyně is also outside the area.

[12] The railway line from Temelín station to Temelín NPP is power plant siding.

In terms of the requirements of criteria defined in Section 5(n) of Decree No. 215/1997 Coll. [L. 1], the siting of the power plant is not conditioned by measures in power plant design.

2.1.7.5 REQUIREMENTS IN PARA. 4.10 TO 4.13 OF THE IAEA STANDARD NS-R-3

The text of the requirements in para. 4.10 to 4.13 of the IAEA standard NS-R-3 [L. 6] is reproduced in Tab. 21 under item 1.5.

In accordance with the requirements stipulated in para. 4.10 to 4.13 of the IAEA standard NS-R-3 [L. 6], information on the number of the population in the area within 100 km from ETE3,4 was collected. The values reported by the Czech Statistical Office as at 31 December 2009 were used for the territory of the Czech Republic. Information published on Internet portals of the municipalities was used for the territories of Germany and Austria. For the population living in the immediate vicinity of ETE3,4 (up to 5 km), the number of the population by residences was indicated separately for attached municipalities; the data used was internal information provided by the elaborator, dated as at 31 March 2004, used for the purposes of the Preoperational Safety Report ETE1,2 [L. 18]. No hospitals, prisons or schools are located in this immediate vicinity, except for schools in the Temelín, Dříteň and Týn nad Vltavou municipalities. The current civic amenities in the circuit of 5 km around NPP Temelín, the so-called on-site emergency planning zone, are mapped in Section 2.1. hereof.

According to the information of the recipient of the Development Principles of the landscape [L. 218], no projects changing the established demographical data are planned in this document.

The survey of the infrastructure performed by the elaborator of the report [L. 31] shows that there are no industrial, recreational and accommodation facilities in the near region of the Temelín site leading to daily or seasonal migration of the population associated with changes in the number of the population significant in terms of the determination of collective effective doses.

In accordance with the requirements stipulated in para. 4.13 of the IAEA standard NS-R-3 [L. 6], the values of residents in the individual residences were converted by their positions into sectors corresponding to a sixteen-point wind rose.

An emergency planning zone of ETE1,2 exists in the vicinity of ETE3,4 with measures to protect the population in case of a potential accident in the nuclear installation. In the mentioned zone with an approximate radius of 13 km, there are 27,000 people permanently registered (the rounded value was indicated with regard to its fluctuation). The survey of external influences shows that there is no facility in the zone, which would significantly increase the indicated values due to employment and stay in recreational, education and hospital facilities.

The area of siting of NPP Temelín is under agricultural cultivation. The Hněvkovice reservoir and the Kořensko plant, both located on the Vltava River, are the nearest water bodies. In the vicinity of NPP Temelín within approximately 10 km, neither protection zones of water sources, nor protected area of natural water accumulation occur.

The use of land and water at Temelín site in connection with the operation of ETE1,2 is under control in terms of the assessment of impact of the nuclear installation on the components of the food chain. The results of the inspection carried out by the operator of ETE1,2 are documented in reports [L. 36 to L. 42].

The preliminary assessment of impact of the operation of the proposed installation on personnel, the public and the environment is the subject of Chapter 4 of this Initial Safety Analysis Report. The evaluation of the potential radiological impacts of accidental releases of radioactive material arising from the safety analyses shall be provided in the Preoperational Safety Report ETE3,4 (Appendix B.I. 2 to Act No. 18/1997 Coll. [L. 2]).

These surveys and information provided in Chapter 4 of this Initial Safety Analysis Report meet the purpose of the investigation defined in para. 4.10 to 4.13 of the IAEA standard NS-R-3 [L. 6].

2.1.7.6 REQUIREMENTS IN PARA. 4.14 OF THE IAEA STANDARD NS-R-3

The text of the requirement in para. 4.14 of the IAEA standard NS-R-3 [L. 6] is reproduced in Tab. 21 under item 1.6.

The characteristics of the use of the area and water sources in the area of NPP Temelín are specified in Sections 2.1.2.2.15, 2.1.2.3.1 and 2.1.2.3.2.

The impact of the operation of the nuclear installation on its vicinity, ground and surface water, including food chains, is monitored by technical facilities of the laboratory for monitoring of environment radioactivity of the operator of ETE1,2 (see Section 2.7.2.3 hereof).

The assessment of impacts on water sources included an analysis of the process of the contaminant spreading in the Vltava cascade and Elbe River, see Section 2.1.2.2.15 hereof. For further hydrological information see Section 2.6.2 hereof.

An overview of drinking water sources for the population in the vicinity of ETE3,4 is indicated in Tab. 11.

An off-site emergency plan for ETE1,2, which is modified as needed, was created and approved for the site.

These surveys meet the purpose of the investigation defined in para. 4.14 of the IAEA standard NS-R-3 [L. 6].

2.1.8 FINAL ASSESSMENT

The performed site surveys and their analyses show that:

- The site proposed for ETE3,4 does not encroach on the protection zones of motorways and railways (Section 4(q) of Decree No. 215/1997 Coll. [L. 1]) and the infrastructure defined in Section 4(l),(m) and (n) of Decree No. 215/1997 Coll. [L. 1]. Siting of ETE3,4 is in compliance with the exclusion and conditional criteria related to the protection zones of other installations.
- Demographical data and area characteristics and use of water in the vicinity of ETE3,4 are available for the ETE3,4 site for the purposes of analysing the impact of the power plant on the population and the environment in accordance with the requirements stipulated in para. 4.10 to 4.14 of the IAEA standard NS-R-3 [L. 6].

No measures resulted for power plant design basis from the assessment of the criteria defined in Decree No. 215/1997 Coll. [L. 1], and the requirements in the IAEA standard NS-R-3 [L. 6] included in Section 2.1.

2.2 CLOSE INDUSTRIAL, TRANSPORTATION AND MILITARY BUILDINGS

2.2.1 SCOPE OF THIS SECTION

The subject of this section is to present the results of the analysis of the Temelín near region characteristics in terms of industrial production, transportation and storage of hazardous materials, which could, under unfavourable circumstances, be a source of risk to nuclear safety of ETE3,4. This concerns industrial, transportation and military facilities, product pipelines, operation of aircraft, sources of electromagnetic radiation (electromagnetic interference) and forest fire in the vicinity. The requirements in the IAEA standard NS-R-3 [L. 6] applied to siting of nuclear installations relate to the problems defined in the standard cited in para. 3.44 to 3.51 as "external human induced events".

The term "external sources" identifies the sources of events, sources of risk and events outside the site area of the Temelín NPP. The term "internal sources" identified the facilities and routes, sources of risk and events inside the site area of ETE1,2; the internal sources are described in Section 2.3. The sources of events are further divided into stationary (associated with a fixed object) and mobile (associated with a moving object). In relation to the aforementioned division of the sources of events, these sources are hereinafter assigned identification numbers with the following prefixes:

ES – external stationary	IS – internal stationary
EM – external mobile	IM – internal mobile

2.2.2 SUMMARY OF FACTS

2.2.2.1 INDUSTRIAL AND MILITARY FACILITIES AND FOREST STANDS

2.2.2.1.1 Identification of Event Sources

In accordance with the report [L. 31], several smaller agricultural plants with no agrochemical storage facilities are operated in the vicinity of the Temelín NPP. The Týn nad Vltavou and Temelín municipalities include small-scale auxiliary manufacturing; the petrol stations in the vicinity, Wienerberger brickworks and Graphite plant in Týn nad Vltavou are the most significant facilities in terms of transportation.

The ČEPS substation is situated southwest of the premises of the Temelín NPP; the area of the small-scale auxiliary manufacturing includes the quarry and the construction of the bio-ethanol production plant is planned at the Býšov site southeast of the premises of the Temelín NPP [L. 31].

According to the report [L. 31], several petrol stations are located in the delineated zone: in Týn nad Vltavou (SHELL, Autoservis Novotný, ČSAD Jihotrans), ATOS in Albrechtice nad Vltavou, Flaga in Chlumec and Š + H oil in Temelín. The nearest petrol station is the Š + H oil in Temelín and the largest tank is at the ČSAD Jihotrans station. The petrol station in Temelín is closest to the Temelín NPP. Other petrol stations will be evaluated by means of the largest and closest to the Temelín NPP – the petrol station in Týn nad Vltavou.

In the vicinity of the Temelín NPP, there are three sites that include chemical facilities. One is situated on the northwest edge of the Temelín municipality, the second one is situated east of the Temelín NPP in the area called Hůrka and the last one is situated in the vicinity of the Mydlovary municipality. The first case involves a plant that was operated by the Chema company in the 1990s. It is currently owned by the SUNEX company, engaged in the collection and sorting of waste in Temelín. A compost facility is identified as a chemical facility in the second case. The third case concerns the MAPE uranium milling plant, which was closed in the 1990s and the premises of which shall be gradually reclaimed.

No military facilities occur within 10 km from the Temelín NPP.

The larger-scale industries are situated farther away.

Facilities identified as external stationary event sources in industrial and military facilities and forest stands and hazardous substances occurring therein are indicated in Tab. 22.

A facility, which conservatively represents all the detected petrol stations, is indicated in the table as a petrol station - it is situated at the place of the nearest station and is equipped with the largest fuel storage tanks.

Tab. 22 External stationary event sources

Identification	Facility	Hazardous substances
ES1	very high voltage switchyard in Kočín	transformer oil
ES2.1	petrol station in Týn nad Vltavou	fuels in underground storage tanks
ES2.2	petrol station in Temelín	fuels in underground storage tanks
ES3	Wienerberger brickworks in Týn nad Vltavou	diesel oil in an overhead storage tank
ES4	Graphite Týn nad Vltavou - graphite product plant	sulphuric acid, sodium hydroxide, hydrogen peroxide
ES5	Slavětice stone quarry	commercial explosives
ES6	bio-ethanol production plant at Býšov	aerial dispersions of flammable dusts, ethanol, petrol, ammonia water, sodium hydroxide, sulphuric acid
ES7	forest stands in the vicinity of Temelín NPP	flammable stand

External stationary event sources identified during data collection are depicted in the layout of sources (Dwg. 3).

The quantity of hazardous substances in the individual sources of events, evaluated as significant for further analysis of stationary event sources, and the distance of the individual sources of events from ETE3,4 are indicated in Tab. 23.

Tab. 23 Description of external stationary risk sources

Identification	Risk source	Description	Location
ES1.1	transformer oil in Kočín switchyard	more than 160 m^3 in total, largest separated volume up to 68 t	2,200 m southwest of the fence of Temelín NPP
ES2.1	fuels in Týn nad Vltavou petrol station	up to 111 t in the station, up to 76.5 t in the largest separated volume	4,900 m northeast of the fence of Temelín NPP
ES2.2	fuels in Temelín petrol station	up to 45 m^3 in the station, up to 25 m^3 in the largest separated volume	700 m northeast of the fence of Temelín NPP
ES3.1	diesel oil in the Wienerberger brickworks	up to 8.5 t in the largest separated volume	4,600 m northeast of the fence of Temelín NPP
ES4.1	sulphuric acid in the Graphite production plant	35 t, concentration 96%	4,900 m northeast of the fence of Temelín NPP
ES5.1	blasting explosives in Slavětice quarry	8 t, equivalent TNT = 1	more than 5,000 m north of the fence of Temelín NPP
ES6.1	dispersion of flammable dusts in Býšov bio-ethanol production plant	up to 18,000 m^3 in the largest separated volume, equivalent TNT = 0.5 kg/m^3	more than 1,700 m southeast of the fence of Temelín NPP
ES6.2	ethanol in Býšov bio-ethanol production plant	up to 5,000 t in total, up to 3,018 t in the largest separated volume	more than 1,700 m southeast of the fence of Temelín NPP
ES6.3	natural gas in Býšov bio-ethanol production plant	approximately 4,000 kg/hr	more than 1,700 m southeast of the fence of Temelín NPP
ES6.4	ammonia water in Býšov bio-ethanol production plant	35 t, concentration 25%	more than 1,700 m southeast of the fence of Temelín NPP
ES6.5	sulphuric acid in Býšov bio-ethanol production plant	36.6 t, concentration 96%	more than 1,700 m southeast of the fence of Temelín NPP
ES7.1	forest stands in the vicinity of Temelín NPP	approximate acreage 0.5 km^2	at the northeast corner of the fence of the Temelín NPP, more than 500 m north of the northwest corner of the fence of the Temelín NPP

2.2.2.1.2 Identification of Events

In accordance with [L. 35] an overview of events, which may occur in the identified sources of events, is indicated in Tab. 24. The events were identified in accordance with the approach to the identification of the sources of events. The following substances of the identified substances are not included in the table. The analysis does not deal with sodium hydroxide solutions as they are solid solutions in water and there is no reason to suppose that significant clouds of toxic substances would form and spread due to the escape of such solutions. The hydrogen peroxide in the Graphite plant was not included as it is present at a low level and the occurrence of a cloud of oxidizing substance capable of jeopardizing the Temelín NPP may be excluded even without calculation. The petrol in the bio-ethanol production plant is neglected due to the multiple quantity of ethanol in the production plant.

Tab. 24 External stationary sources of events and their effects

Identification	Facility	Hazardous substances	Event types
ES1	switchyard in Kočín	transformer oil	fire
ES2	petrol station in Temelín	fuels	fire, explosion, missiles, creation and spreading of flammable substances
ES3	Wienerberger brickworks in Týn nad Vltavou	diesel oil	fire, explosion, missiles, creation and spreading of flammable substances
ES4	graphite plant in Týn nad Vltavou	sulphuric acid	spreading of toxic substance clouds
ES5	Slavětice stone quarry	commercial explosives	explosion, missiles
ES6	bio-ethanol production plant at Býšov	air dispersions of flammable dusts	explosion, missiles
		ethanol	fire, explosion, missiles, creation and spreading of flammable substances
		natural gas	fire, explosion, missiles, creation and spreading of flammable substances
		ammonia water	spreading of toxic substance clouds
		sulphuric acid	spreading of toxic substance clouds
ES7	forest stands in the vicinity of Temelín NPP	flammable stand	fire

The sources of events and risks are described in detail in the report [L. 35].

2.2.2.1.3 Evaluation of Events

The analysis of interactions between the external stationary risk sources and ETE3,4 is described in Section 6.1.4 of the report [L. 35]. In accordance with the method described in Section 2.2.5.1 hereof, the radiological effects of events were first assessed and in the cases when they cannot be neglected the frequency of interaction was also determined. The results of the preliminary assessment of the effects of interaction and frequencies of interaction of ETE3,4 and events associated with external stationary risk sources are indicated in Tab. 25.

Tab. 25 Preliminary assessment results in external stationary event sources

Source identification	Risk source name	Event type	Interaction effect	Interaction frequency
ES1.1	transformer oil in Kočín switchyard	fire	can be neglected	---
ES2.1	fuels in Týn nad Vltavou petrol station	fire	can be neglected	---
		spreading of an explosive cloud	can be neglected	---
		explosion of a cloud	can be neglected	---
ES2.2	fuels in Temelín petrol station	fire	can be neglected	---
		spreading of an explosive cloud	cannot be neglected	cannot be neglected
		explosion of a cloud	can be neglected	---
ES3.1	diesel oil in the Wienerberger brickworks	fire	can be neglected	---
		spreading of an explosive cloud	can be neglected	---
		explosion of a cloud	can be neglected	---
ES4.1	sulphuric acid in the Graphite production plant	spreading of a toxic cloud	cannot be neglected	cannot be neglected
ES5.1	blasting explosives in Slavětice quarry	explosion of an explosive	can be neglected	---
ES6.1	dispersion of flammable dusts in Býšov bio-ethanol production plant	explosion of an explosive	can be neglected	---
ES6.2	ethanol in Býšov bio-ethanol production plant	fire	can be neglected	---
		spreading of an explosive cloud	cannot be neglected	cannot be neglected

Source identification	Risk source name	Event type	Interaction effect	Interaction frequency
		explosion of a cloud	cannot be neglected	cannot be neglected
ES6.3	natural gas in Býšov bio-ethanol production plant	fire of escaping gas	can be neglected	---
		spreading of an explosive cloud	can be neglected	---
		explosion of a cloud	can be neglected	---
ES6.4	ammonia water in Býšov bio-ethanol production plant	spreading of a toxic cloud	can be neglected	---
ES6.5	sulphuric acid in Býšov bio-ethanol production plant	spreading of a toxic cloud	cannot be neglected	cannot be neglected
ES7.1	forest stands in the vicinity of Temelín NPP	fire	cannot be neglected	cannot be neglected

The preliminary assessment of the effects of interactions of events with risk sources ES2.2, ES4.1, ES6.2, ES6.5, ES7.1 and ETE3,4 shows that they cannot be neglected.

The detailed assessment of the effects and frequencies of interactions described in Section 6.1.5 of the report [L. 35] shows that the effects of interactions of events with risk sources ES2.2, ES4.1, ES6.2, ES6.5, ES7.1 at ETE3,4 can be neglected.

2.2.2.2 RAIL TRANSPORT

2.2.2.2.1 Identification of Event Sources

According to the report [L. 31], three line sections of railway lines operated by České dráhy, line Nos. 190, 192 and 200, are located in the vicinity of Temelín NPP. Only one line, specifically line No. 192, encroaches closer than 10 km from the premises of Temelín NPP.

Line No. 192 is of local significance. It is used both for passenger and for freight transport. A railway siding branches off from the public line No. 192 in the Temelín station to the Temelín NPP site area. The dangerous goods are permitted to be transported by line no. 192 including railway siding to the Temelín NPP. The operator of the line provided brief statistics of transported types of dangerous goods. The following four items were selected from the table as substances that should be assessed for potential interaction with Temelín NPP: ammonium, ammonium nitrate, nitric acid and sulphuric acid. These substances should include two more substances transported by rail to the Temelín NPP: diesel oil and ammonia water.

The railway routes indicated in Tab. 26 were identified as external mobile sources of events associated with rail transport.

Tab. 26 Railway lines assessed as external sources of events

Identificat ion	Route	Hazardous substances
EM1	public railway line Číčenice - Týn nad Vltavou, section Temelín - Bohunice	ammonium, ammonium nitrate
EM2	railway siding Temelín - Temelín NPP	ammonia water, nitrogen acid, sulphuric acid, diesel oil

2.2.2.2.2 Identification of Events

With regard to the hazardous substances transported on such lines, the types of events listed in Tab. 27 were assessed.

Tab. 27 Event types considered in railway external mobile event sources

Identifica tion	Event source	Substances	Event types
EM1	public railway line	ammonia	spreading of toxic substance clouds
		ammonium nitrate	explosion, missiles
EM2	railway siding	ammonia water, nitric acid, sulphuric acid	spreading of toxic substance clouds
		diesel oil	fire, explosion, missiles, creation and spreading of flammable substances

In addition to the amount of hazardous substance, the frequency of transportation should be taken into consideration to assess the risk associated with mobile event sources on railway siding. For transportation of hazardous substances by railway siding, the increased numbers of transportation are expected compared to the situation analysed in 2004 [L. 18], in the ratio 5:2, because the construction of ETE3,4 will increase the volume of this type of transportation. An overview of this data is shown in Tab. 28.

Tab. 28 Risk sources in railway external mobile event sources

Identification	Risk source	Description	Location
EM1.1	ammonia on a public railway line	25 t to 12/year	1,100 m north of the fence of Temelín NPP
EM1.2	ammonium nitrate on a public railway line	50 t to 5/year	1,100 m north of the fence of Temelín NPP
EM2.1	sulphuric acid on railway siding	50 t, concentration 96%, 5 times a year	at the northwest corner of the fence of the Temelín NPP
EM2.2	nitric acid on railway siding	25 t in 40 kg barrels, concentration 65%, 5 times a year	at the northwest corner of the fence of the Temelín NPP
EM2.3	ammonia water on railway siding	25 t, concentration 25%, 5 times a year	at the northwest corner of the fence of the Temelín NPP
EM2.4	diesel oil on railway siding	up to 300 m^3 in total, largest separated volume 50 m^3, 3 times a year	at the northwest corner of the fence of the Temelín NPP

2.2.2.2.3 Evaluation of Events

The preliminary assessment of the effects and frequencies of interactions of risks resulting from mobile sources associated with railway transport shows that the events listed in Tab. 29 cannot be neglected.

Tab. 29 Events in railway external mobile sources, which cannot be neglected based on the preliminary assessment

Source identification	Risk source name	Event type	Interaction effect	Interaction frequency
EM1.1	ammonia on a public railway line	spreading of a toxic cloud	cannot be neglected	cannot be neglected
EM2.1	sulphuric acid on railway siding	spreading of a toxic cloud	cannot be neglected	cannot be neglected
EM2.2	nitric acid on railway siding	spreading of a toxic cloud	cannot be neglected	cannot be neglected
EM2.3	ammonia water on railway siding	spreading of a toxic cloud	cannot be neglected	cannot be neglected
EM2.4	diesel oil on railway siding	liquid fire	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	

The detailed assessment of the effect and frequency of interactions (see Section 6.2.5 of the report [L. 35]) shows that the following two risks associated with the event of type "spreading of a toxic cloud" cannot be neglected.

EM2.2 NITRIC ACID ON RAILWAY SIDING

Identification of events. Only one event is analysed: spilling of the acid, its evaporation and the spreading of a cloud of acid.

Creation and spreading of a toxic cloud The extreme and with regards to the package of acid also very improbable event is the spilling of all acid in the truck, evaporation from the resulting puddle and spreading of toxic vapours. Twenty five tons of 65% acid has a volume of approximately 18.8 m^3 and could be spilled on the area of approximately 940 m^2. The equivalent diameter of the puddle would be approximately 35 m. Based on available data, the tension of 65% acid vapours (UN 2031) at 311 K (38°C) up to 25 mm Hg, i.e. approximately 3300 Pa. The calculation IM1.2-1 shows that at a wind speed of 1 m/s the amount of evaporated acid would be up to 0.11 kg/s. The acid vapours are heavier than air. Calculation IM1.2-2, where the acid vapour is considered a heavy gas and admixing rate of oxygen above the puddle is included in the calculation, shows that the reach of estimated concentration ERPG-2 at the same wind speed and under the worst day-time stability could be approximately at most 1,100 m. Whereas the supply lines of the control room ventilation system can be at an approximate distance of 470 m, the effects of the interaction of the event and the power plant cannot be neglected.

It is possible to evaluate the frequency based on an estimation of line section length along buildings with items important for safety in km, frequency of passages of sets with acid on the line per year and estimation of accident probability during the passage of a single set on 1 km of the line. The section along buildings with air intakes equals approximately 2.4 km (from the railway gate to the place in front of the building 631/01 and from here to the public railway line beyond the distance of critical reach). Evaluation of event frequency is therefore $2.4 \times 5 \times 5 \times 10^{-7} = 6 \times 10^{-6}$[1/year]. If we consider that the leak creation is estimated for one out of four accidents (which is probably too strict for transportation in barrels) and the probability of spreading the cloud towards buildings with items important to safety does not exceed at any location approximately 0.38, the frequency can be evaluated as 5.7×10^{-7}[1/year]. Therefore this event cannot be neglected even with regards to the frequency.

EM2.3 AMMONIA WATER ON RAILWAY SIDING

Identification of events. Only one event is analysed: spilling of ammonia water, its evaporation and the spreading of a cloud of ammonia.

Creation and spreading of a toxic cloud The extreme event would be the spilling of the whole cistern content, evaporation from the resulting puddle and spreading of toxic vapours. Twenty five tons of the liquid would have a volume of approximately 22 m^3 and could be spilled on the area of approximately 1,100 m^2. The equivalent diameter of the puddle would be approximately 37 m. Based on data in the safety sheet, the tension of vapours of the liquid at 293 K (20°C) reaches approximately 64,000 Pa. Calculation IM1.3-1 shows that at a wind speed of 1 m/s the amount evaporated from the puddle would be 1.11 kg/s at most. Calculation IM1.3-2 (source modelled as aerial) shows the reach of ERPG-2 concentration at the same wind speed and under the worst day-time stability could be at most 770 m. Whereas the supply lines of the control room ventilation system can be at an approximate distance of 470 m, the effects of the interaction of the event and the power plant cannot be neglected.

It is possible to evaluate the frequency based on an estimation of line section length along buildings with items important for safety in km, frequency of passages of sets with acid on the line per year and estimation of accident probability during the passage of a single set on 1 km of the line. The section along buildings with air intakes equals approximately 2.3 km (from the railway gate to the place in front of the building 631/01 and from here to the public railway line beyond the distance of critical reach). Evaluation of event frequency is therefore $2.3 \times 5 \times 5{\times}10^{-7} = = 5.75{\times}10^{-6}$/year. If we consider that the leak creation is estimated for one out of four accidents and the probability of spreading the cloud towards buildings with items important to safety does not exceed at any location approximately 0.38, the frequency can be evaluated as $5.46{\times}10^{-7}$/year. Therefore this event cannot be neglected even with regards to the frequency.

Based on the analysis performed, the events EM2.2 and EM2.3 (nitric acid and ammonia water on railway siding) should be included among design events and the resistance to such events should be included in design basis arising from the site assessment in accordance with the siting criteria of a nuclear installation listed in Section 5 of Decree No. 215/1997 Coll. [L. 1], and the requirements stipulated in the IAEA standard NS-R-3 [L. 6]. The attended areas, which require the presence of operating personnel to assure the nuclear safety, should be provided with technical equipment and trained procedures to protect the operating personnel. Specifically, control rooms should be provided with such equipment and procedures.

2.2.2.3 ROAD TRANSPORT

2.2.2.3.1 Identification of Event Sources

There are a total of five secondary roads within 5 km around the Temelín NPP site. The roads II/105, II/138 and II/141 are nearest the Temelín NPP site. Accidents and incidents, which happen on these roads in the vicinity of the Temelín NPP, have the highest potential to affect the items important to the nuclear safety. More details collected in the reports [L. 31] as well as the present analysis focus on the above three roads.

The stretch of road II/105, which is nearest the Temelín NPP site, is the section from the intersection of this road and the road II/138 to the intersection with the branch leading from the road II/141 near Vysoký Hrádek. As to the road II/138, its stretch from Temelín to the intersection with the road II/105 via Temelínec borders with the Temelín NPP site. And finally, from the road II/141 perspective, its stretch from Temelín to the branch leading to Vysoký Hrádek is nearest the Temelín NPP site.

The data used is based on the Road Transport Census conducted in the road network in the vicinity of the Temelín NPP. The figures show that approximately 5,500 vehicles a day drive through the road II/105 stretch, while approx. 700 vehicles a day, being eight times less, drive through the road II/138 stretch. Around 1,400 vehicles a day drive through the road II/141 stretch, i.e. four times less. However, the representation of heavy-duty vehicles, i.e. those preferentially expected to transport the hazardous loads, is different. Over 1,000 vehicles a day drive through the road II/105, 420 vehicles a day through the road II/141, i.e. more than one third, and about

350 vehicles a day drive through the road II/138, i.e. a rough third. We suppose that the ratio of numbers of vehicles representing the risk sources for the Temelín NPP is closer to the heavy-duty figures than the overall numbers.

The report [L. 31] also sums up the accident rate data in the above stretches. Based on the above data, the frequency of road accidents is estimated in the methodological part of this analysis. The reports, however, mention nothing on the nature of loads transported on these roads.

The data contained in the report [L. 30] shows that a total of 13 road tankers with bio-ethanol, each containing more than 13 cubic metres of the product, will be dispatched, on a daily basis, from the bio-ethanol production plant in Býšov. Road tankers will drive along the reconstructed and partially newly built-up road to the intersection with the road II/105, east of Kočín, and from there towards České Budějovice. They will approach the Temelín NPP fencing at a maximum distance of about 1,300 metres. Auxiliary substances are expected to be transported on the road II/105 along the Temelín NPP fence. The maximum volumes of ammonia water and sulphuric acid transported by one vehicle are estimated at 15 t and 5 t, respectively.

The road routes in the Temelín near region, on which the hazardous substances may be transported, are indicated in Tab. 30.

Tab. 30 Road routes with external sources of mobile events

Identification	Route	Hazardous substances
EM3	road II/105 along the southwest side of the site area of the Temelín NPP	commercial explosives, fuels, liquefied petroleum gases (LPG) in tanks and small containers, acetylene in cylinders, acetylene welding set, ammonium nitrate, ammonia water, sulphuric acid, ethanol
EM4	road II/138 along the southeast side of the site area of the Temelín NPP	commercial explosives, fuels, liquefied petroleum gases (LPG) in tanks and small containers, acetylene in cylinders, acetylene welding set, ammonium nitrate
EM5	road II/141 along the southwest side of the site area of the Temelín NPP	commercial explosives, fuels, liquefied petroleum gases (LPG) in tanks and small containers, acetylene in cylinders, acetylene welding set, ammonium nitrate

2.2.2.3.2 Identification of Events

Types of events that can result from road transport are summarized in Tab. 31.

Tab. 31 Types of events associated with road traffic

Identification	Event source	Substances	Event types
EM3	road II/105	commercial explosives	explosion, missiles
		fuels	fire, explosion, missiles, creation and spreading of flammable substances

Identification	Event source	Substances	Event types
		liquefied flammable gases	fire, explosion, missiles, creation and spreading of flammable substances
		flammable gases	fire, explosion, missiles, creation and spreading of flammable substances
		ammonium nitrate	explosion, missiles
		ammonia water	spreading of toxic substance clouds
		sulphuric acid	spreading of toxic substance clouds
		ethanol	fire, explosion, missiles, creation and spreading of flammable substances
EM4	road II/138	commercial explosives	explosion, missiles
		fuels	fire, explosion, missiles, creation and spreading of flammable substances
		liquefied flammable gases	fire, explosion, missiles, creation and spreading of flammable substances
		flammable gases	fire, explosion, missiles, creation and spreading of flammable substances
		ammonium nitrate	explosion, missiles
EM5	road II/141	commercial explosives	explosion, missiles
		fuels	fire, explosion, missiles, creation and spreading of flammable substances
		liquefied flammable gases	fire, explosion, missiles, creation and spreading of flammable substances
		flammable gases	fire, explosion, missiles, creation and spreading of flammable substances
		ammonium nitrate	explosion, missiles

The description of events used for the assessment of risks resulting from external mobile sources of events on road is shown in Tab. 32.

Tab. 32 Risk sources in road external mobile event sources

Identification	Risk source	Description	Location
EM3.1	commercial explosives on road II/105	3 t approximately 12 times a year	20 m southeast of the fence of Temelín NPP

Identification	Risk source	Description	Location
EM3.2	fuels on road II/105	18,000 to 33,000 l up to 2,000 times a year	20 m southeast of the fence of Temelín NPP
EM3.3	LPG in a tank on road II/105	10 t up to 600 times a year	20 m southeast of the fence of Temelín NPP
EM3.4	LPG in small container on road II/105	up to 100 pcs by 33 kg up to 6,000 times a year	20 m southeast of the fence of Temelín NPP
EM3.5	acetylene in cylinders on road II/105	up to 10 pcs by 10 kg up to 200 times a year	20 m southeast of the fence of Temelín NPP
EM3.6	acetylene welding set on road II/105	One 10 kg unit up to 20,000 times a year	20 m southeast of the fence of Temelín NPP
EM3.7	ammonium nitrate on road II/105	up to 15 t up to 200 times a year	20 m southeast of the fence of Temelín NPP
EM3.8	ammonia water on road II/105	up to 15 t up to 50 times a year	20 m southeast of the fence of Temelín NPP
EM3.9	sulphuric acid on road II/105	up to 5 t up to 30 times a year	20 m southeast of the fence of Temelín NPP
EM3.10	ethanol on road II/105	up to 15 m^3 up to 4,800 times a year	20 m southeast of the fence of Temelín NPP
EM4.1	commercial explosives on road II/138	3 t approximately 3 times a year	220 m southwest of the fence of Temelín NPP
EM4.2	fuels on road II/138	18,000 to 33,000 l up to 500 times a year	220 m southwest of the fence of Temelín NPP
EM4.3	LPG in a tank on road II/138	10 t up to 150 times a year	220 m southwest of the fence of Temelín NPP
EM4.4	LPG in small container on road II/138	up to 100 pcs at 33 kg each up to 1,500 times a year	220 m southwest of the fence of Temelín NPP
EM4.5	acetylene in cylinders on road II/138	up to 10 pcs at 10 kg each up to 50 times a year	220 m southwest of the fence of Temelín NPP
EM4.6	acetylene welding set on road II/138	1 pc at 10 kg up to 5,000 times a year	220 m southwest of the fence of Temelín NPP
EM4.7	ammonium nitrate on road II/138	up to 15 t up to 50 times a year	220 m southwest of the fence of Temelín NPP
EM5.1	commercial explosives on road II/141	3 t approximately 4 times a year	180 m northeast of the fence of Temelín NPP
EM5.2	fuels on road II/141	18,000 to 33,000 l up to 670 times a year	180 m northeast of the fence of Temelín NPP
EM5.3	LPG in a tank on road II/141	10 t up to 200 times a year	180 m northeast of the fence of Temelín NPP
EM5.4	LPG in small container on road II/141	up to 100 pcs by 33 kg up to 2,000 times a year	180 m northeast of the fence of Temelín NPP
EM5.5	acetylene in cylinders on road II/141	up to 10 pcs by 10 kg up to 67 times a year	180 m northeast of the fence of Temelín NPP

Identification	Risk source	Description	Location
EM5.6	acetylene welding set on road II/141	One 10 kg unit up to 6,700 times a year	180 m northeast of the fence of Temelín NPP
EM5.7	ammonium nitrate on road II/141	up to 15 t up to 67 times a year	180 m northeast of the fence of Temelín NPP

2.2.2.3.3 Evaluation of Events

The preliminary assessment of effects and frequencies of Temelín NPP interactions with events associated with road external sources of risk described in Section 6.4.3 of the report [L. 35] shows that the events listed in Tab. 33 require more detailed analysis.

Tab. 33 External road sources of risk requiring more detailed analysis

Source identification	Risk source name	Event type	Interaction effect	Interaction frequency
EM3.2	fuels on road II/105	liquid fire	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	
EM3.3	LPG in a tank on road II/105	fire of liquefied gas	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	
EM3.4	LPG in small container on road II/105	fire of liquefied gas	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	
EM3.8	ammonia water on road II/105	spreading of toxic substance clouds	cannot be neglected	cannot be neglected
EM3.9	sulphuric acid on road II/105	spreading of toxic substance clouds	cannot be neglected	cannot be neglected
EM3.10	ethanol on road II/105	fire	can be neglected	cannot be neglected
		spreading of flammable clouds	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	

Source identification	Risk source name	Event type	Interaction effect	Interaction frequency
EM4.2	fuels on road II/138	liquid fire	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	
EM4.3	LPG in a tank on road II/138	fire of liquefied gas	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	
EM4.4	LPG in small container on road II/138	fire of liquefied gas	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	
EM4.5	acetylene in cylinders on road II/138	spreading of a flammable cloud	cannot be neglected	cannot be neglected
		explosion of a flammable cloud	cannot be neglected	
EM4.7	ammonium nitrate on road II/138	explosion of an explosive	cannot be neglected	cannot be neglected
EM5.2	fuels on road II/141	liquid fire	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	
EM5.3	LPG in a tank on road II/141	fire of liquefied gas	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	
EM5.4	LPG in small container on road II/141	fire of liquefied gas	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	cannot be neglected	

According to the more detailed analysis of the effect of events listed in the report [L. 35], the impacts of all events on the power plant can be neglected, except for the event "EM3.8 AMMONIA WATER ON ROAD II/105".

Event EM3.8 Ammonia Water on Road II/105

Since ammonia is a toxic gas and ammonia water is a corrosive and toxic liquid, only one event is expected: creation and spreading of toxic substance clouds

Creation and spreading of a toxic cloud The extreme event would be the spilling of the whole cistern content, evaporation from the resulting puddle and spreading of toxic vapours. Fifteen tons of the liquid would have a volume of approximately 13.2 m^3 and could be spilled on the area of approximately 2,640 m^2. The equivalent diameter of the puddle would be approximately 58 m. Based on data in the safety sheet, the tension of vapours of the liquid at 293 K (20°C) reaches approximately 64,000 Pa. Calculation EM3.8-1, documented in the report [L. 35], shows that at a wind speed of 1 m/s the amount evaporated from the puddle would be 2.60 kg/s at most. Calculation EM3.8-2 (source modelled as aerial) shows the reach of ERPG-2 concentration at the same wind speed and under the worst day-time stability could be at most 1,100 m. Whereas the supply lines of the control room ventilation system can be at an approximate distance of 1,050 m, the effects of the interaction of the event and ETE3,4 cannot be neglected.

It is possible to evaluate the frequency based on an estimation of line section length along buildings with items important for safety in km, frequency of passages of sets with ammonia water on the line per year and estimation of accident probability during the passage of a single set on 1 km of the line. Evaluation of event frequency is therefore $3.2 \times 50 \times 3.3 \times 10^{-6} = 5.28\times10^{-4}$ [1/year]. Therefore this event cannot be neglected even with regards to the frequency.

Whereas the interactions between the event EM3.8 cannot be neglected, the spreading of toxic clouds of ammonia from road II/105 should be included among design events and the resistance to such event should be included in design basis arising from the site assessment in accordance with the siting criteria of a nuclear installation listed in Section 5 of Decree No. 215/1997 Coll. [L. 1], and the requirements stipulated in the IAEA standard NS-R-3 [L. 6]. Similar requirement for design basis is already defined in conclusions of Section 2.2.2.2.3 hereof (to the analysis of events EM2.2 and EM2.3 - nitric acid and ammonia water on railway siding).

2.2.2.4 AIR TRAFFIC

2.2.2.4.1 Identification of Event Sources

2.2.2.4.1.1 Airspace Design

The regular flight operation is organized, as far as possible (given by the possibilities of the Air Traffic Control Centre in the given area), along the regular airways. The flight altitudes are given in flight levels (FL), i.e. altitudes expressed as hundreds of feet (ft), corresponding to barometric altitudes measured under the so-called standard atmosphere conditions (pressure on the sea level: 1,013.25 hPa, zero altitude temperature: 15°C). FL100 equals 10,000 ft, i.e. 3,048 m [L. 265].

Upper Airspace

The upper airspace extends from the flight level FL245 (24,500 ft – 7,468 m) to FL660 (66,000 ft, 20,117 m). Air routes in the upper airspace are predominantly designed for the transit traffic. The highest traffic density concentrates between FL300 and FL400, i.e. between 9,150 and 12,200 metres. Traffic on these levels does not conflict in any way whatsoever with the protective area around the ETE1,2.

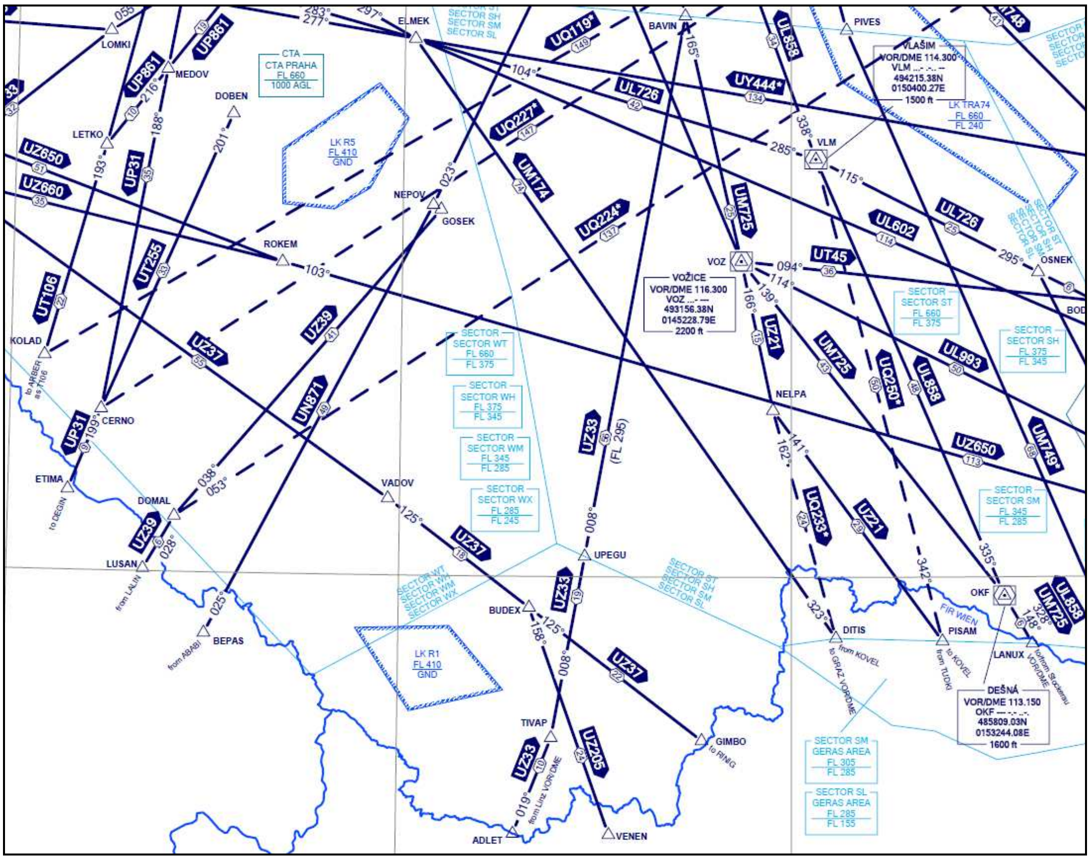

Fig. 5 Airways in the upper airspace as of April 2012.

Lower Airspace

The lower airspace extends from the ground to FL245 (24,500 ft – 7,468 m) and includes an uncontrolled airspace of the G class between the ground and the altitude of 1,000 ft above the ground, where no IFR traffic is permitted and, as a result, no regular air transport may be operated. The maximum allowable speed set for the airspace between the ground and FL100 is 250 kt (1 knot = 1 nautical mile (NM) per hour, 1 NM=1,852 m), i.e. maximum of 463 km/h. The most of the general aviation (GA) flights are operated within this airspace, especially its lower part, i.e. up to 6,000 ft (1,830 m).

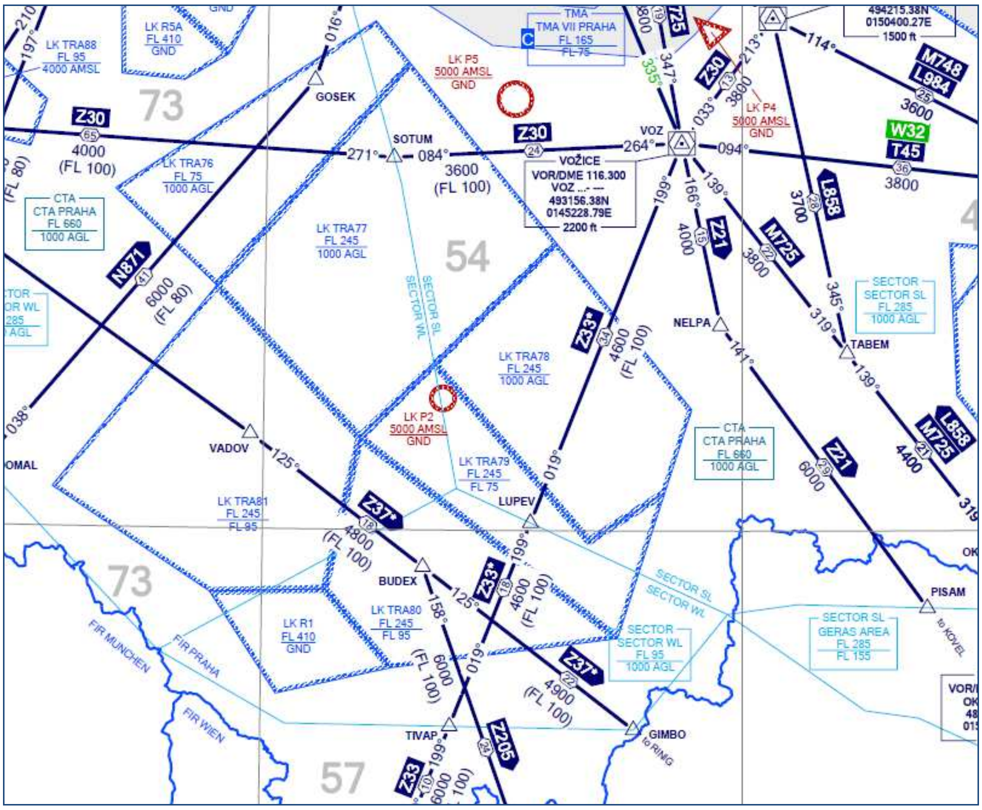

Fig. 6 Airways in the lower airspace as of April 2012

2.2.2.4.1.2 Airspace in the Vicinity of Temelín NPP Site

There is an LK P2 prohibited area around ETE1,2 with a radius of 2 km (1.1 NM), with the centre at:

49 10 48,73 N 014 22 13,77 E

The prohibited air traffic area (except of rescue, police or military flights) reaches from the ground to the height of 5,000 ft above mean sea level (approximately 1,500 m above sea level).

In 2010, the airspace was reorganized to cancel the restricted area LK R8 with a radius of 22 km reaching from the ground to FL95 (9,500 ft above mean sea level). It was an administrative measure where access to LK R8 area was made subject to notification of the Air Traffic Control Station in Prague. Initially, the measure was declared as additional protection of the nuclear installation against deliberate attack following the 11 September 2001 events in New York. The above measure was cancelled in result of risk reassessment of the airspace protection of the Temelín NPP within the competence of the state.

2.2.2.4.1.3 Airspace Management Around and Above LKP 2 Temelín

The vertical division above the territory of the Czech Republic includes G, E, C[13] areas. The air navigation principles in the above areas are listed in the Czech Aeronautical Information Publication and regularly updated.

The space of class G reaches from the ground to the height of 300 m / 1,000 ft above ground, class E reaches from the upper boundary of class G to FL 95 (2,900 m above sea level), and the area of class C reaches from the upper boundary of class E (FL95) to (typically) FL660.

In addition to prohibited areas (LK P2 area for the Temelín near region), restricted or danger areas are shown in the aeronautical charts. Danger areas are marked LK D and they typically include gas pipeline compression stations where gas is released into the atmosphere under specific conditions. LKR areas (restricted airspaces) have the so-called activation mode with time limited pass during activation. Passing is only possible if there is ground-air-ground radio connection with the relevant air traffic control station that may give clearance to pass. Furthermore, restricted areas of TRA and TSA type are distinguished. In both cases, flying across these areas is only possible when they are not activated and their activation is declared in advance by so-called AUP report. Flying across the TRA area is possible during activation if cleared by the relevant ATC station. Flying across the TSA area during activation is not possible under any circumstances. In the vicinity of the Temelín NPP, there are several restricted areas of TRA, TSA type used for military training purposes.

The original structure of airspace management in the vicinity of Temelín NPP site was revised as of April 2012 to cancel the two-tier structure where airspaces were partly overlapping. The particular areas are activated based on the nature of military flights in the area. In case of the LK TRA 78 area, its upper or lower part can be activated separately. The above mentioned LK TRA79 area covers the actual power plant reaching from FL75 to FL240.

[13] The space type D is not used in the vertical division of the Czech airspace at the time of the preparation of this report. It only applies to CTR and TMA of some airports.

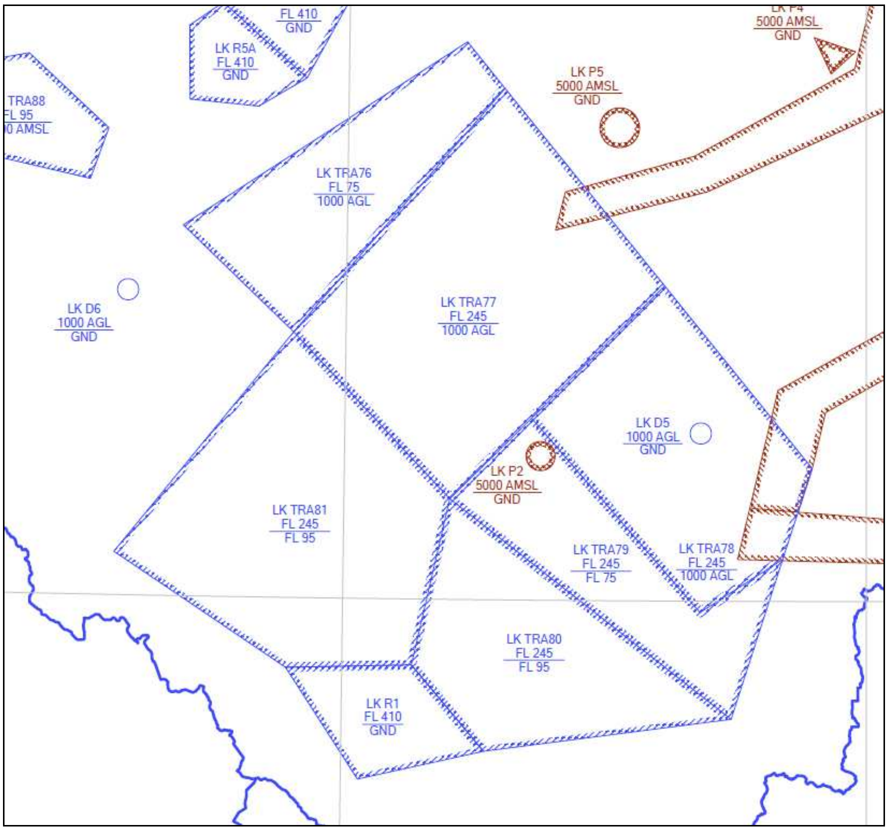

Fig. 7 Layout of airspace boundaries in the vicinity of Temelín NPP

The layout of TRA 78 and TRA 79 areas overlapping according to height in the previous arrangement until April 2012, and confusingly read in the map when just one of them was partly activated (causing their disturbance), was changed. The TRA78 area is now split in the vertical direction into two parts with equal horizontal plan and different planned activation time. The TRA79 area no longer overlaps with TRA78, but is alongside it.

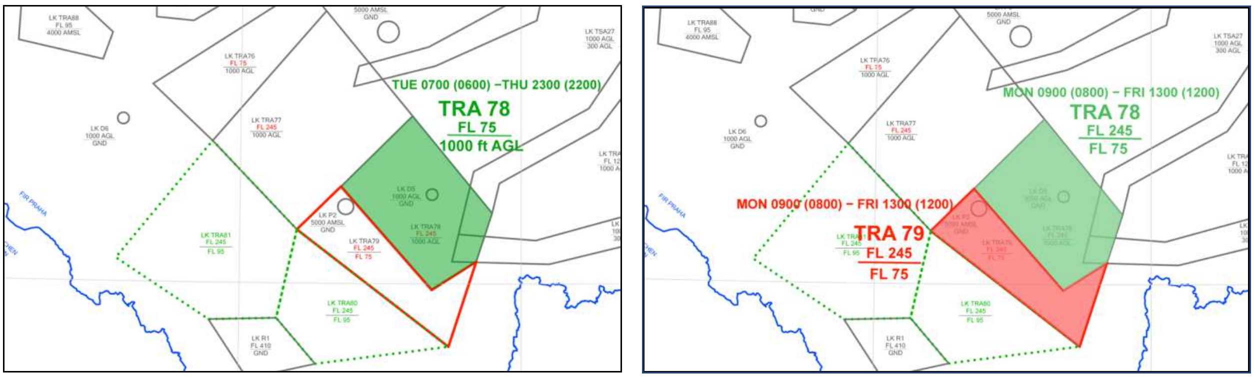

Fig. 8 Layout of TRA78 and TRA79 area boundaries in the vicinity of Temelín NPP

2.2.2.4.1.4 Information for Evaluating Hazards due to Aircraft Accidents

In the vicinity plan of Temelín NPP (Fig. 9), airports (UCL certified) are highlighted in red, verified and non-verified SLZ areas (LAA registered) are shown in blue, and green rectangles with green Roman numerals indicate emergency sites (former agrochemical sites, no longer used). The airports are marked with ICAO codes, their topography and basic parameters are listed in Tab. 34 [L. 265]. It was verified by the Civil Aviation Authority that the list provided in this table is full and complete [L. 200].

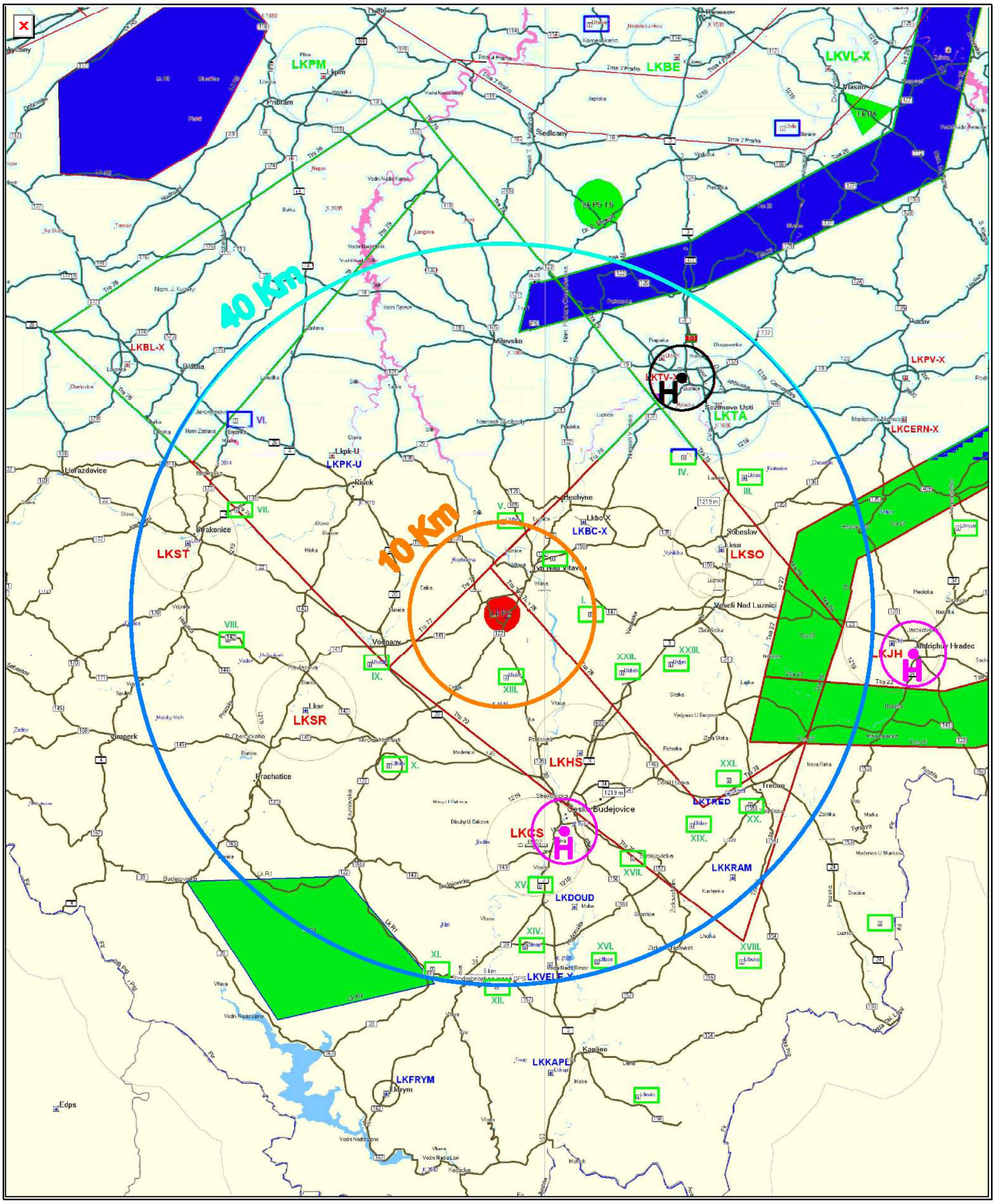

Fig. 9 Map of airports, air recreational vehicle areas and emergency landing sites in the vicinity of the Temelín NPP

Tab. 34 Overview of airports in the vicinity of the Temelín NPP

Name	ICAO code	Coordinates		Runway	Runway orientation	Runway surface
		Longitude	Latitude			
České Budějovice Budějovice	LKCS	N48 56 47	E014 25 39	2,500 m	27/09	Concrete
Hosín	LKHS	N49 02 20	E014 29 15	1,000 m/800 m	06/24	Grass/Asphalt
Soběslav	LKSO	N49 14 48	E014 42 49	740	18/36	Grass
Strakonice	LKST	N49 15 08	E013 53 45	900 m/780 m	03/21,13/31	Grass
Strunkovice	LKSR	N49 04 59	E014 04 33	900 m	15/33	Grass
Tábor	LKTA	N49 23 28	E014 42 30	1,100 m/850 m	12/30,16/34	Grass
Jindřichův Hradec	LKJH	N49 09 03	E014 58 18	700 m/760 m	07/25	Asphalt/Grass
České Budějovice Budějovice., nemocnice	heliport	N48 57 45	E14 27 50	circle ø 10 m		Bitumen
Tábor - hospital	heliport	N49 24 59	E014 39 14	15x15 m		Bitumen
Jindř. Hradec, hospital	heliport	N49 08 21	E015 00 20	circle ø 20 m		Concrete
Temelín parking lot	heliport	**location see** Fig. 10		20x20 m		

Fig. 10 Location of heliport on Temelín NPP parking lot

Other characteristics of airports, specifications of air routes and the affected air spaces are shown in Section "Airports and Other Air Spaces in the Vicinity of the Temelín NPP" of the report on their surveys [L. 34].

The following Sections 2.2.2.4.1.5 to 2.2.2.4.1.12 shows the basic parameters, characteristics of use and possible plans of extension for the individual airports. All cases involve the GA airports, certified by the Civil Aviation Authority. Information was taken from the above cited report on surveys [L. 34].

2.2.2.4.1.5 České Budějovice – LKCS

Status	**Public domestic, non-public international airport**
Operator	Jihočeské letiště Č. Budějovice s.r.o., phone: 386 325 339
Coordinates	N48°56'47" E014°25'39"
Runway, dimensions and surface	09/27, 2,500x45 m, concrete
Frequency	INFO 135,925

The airport is currently operated as uncontrolled, with VFR DAY traffic.

The airport traffic is monitored in three weight groups. Ultralight aircraft with maximum take-off weight up to 450 kg, then aircraft with maximum take-off weight up to 2,000 kg and aircraft with maximum take-off weight more than 2,000 kg.

Tab. 35 Frequency of movements at the České Budějovice Airport

Total number of movements on LKCS						
	2006	**2007**	**2008**	**2009**	**2010**	**2011**
Ultralight aircraft	2,418	5,462	11,282	7,750	7,156	5,386
< 2,000 kg	0	442	678	486	886	687
> 2,000 kg	0	178	178	210	82	83

The operator (Jihočeské letiště company) plans to equip the airport to accept medium cargo and commercial aircrafts (BizJet). The plan of the operator includes the recovery of terminal control area and control zone (TMA and CTR), the provision of the airport with basic radio-navigation facilities for IFR approach and equipment for night traffic. The plan of the airport operator is supported by the Development Principles of the South Bohemian Region [L. 218], where the development of the airport for civil transportation is also taken into account.

2.2.2.4.1.6 Hosín – LKHS

Status	**Public domestic, non-public international airport**
Operator	České Budějovice Aero Club, phone: 387 220 716
Coordinates	N49°02'24" E014°29'42"
Runway, dimensions and surface	06L/24R, 800x24 m, asphalt 06R/24L, 1000x50 m, grass
Frequency	INFO 130,200

The Hosín aircraft is intended for VFR DAY and VFR NIGHT traffic; the airport has no radio equipment for instrument approach and the installation of such radio-navigation equipment is not taken into account in the future. This is an uncontrolled airport, providing only Aerodrome Flight Information Service (AFIS). The runway is suitable for the operation of aircrafts with maximum take-off weight up to 6,500 kg. The change in runway parameters (extension, increase in bearing capacity) is not taken into account in the future because eventual heavier traffic will be handled by nearby České Budějovice LKCS Airport, situated on the south side of the city.

From the monitored airports in the vicinity of the Temelín NPP, the Hosín airport traffic has the highest number of movements, the share of gliders (combined with GA aircrafts in one column) is at least 25-30% in the GA and GLD category due to a long tradition of various glider competitions and concentration at the Hosín airport.

Tab. 36 Frequency of movements at the Hosín airport

Year	GA and GLD domestic	GA international	ULL	Total
2006	8,251	311	3,128	11,690
2007	9,865	493	4,975	15,333
2009	12,656	810	6,299	19,765
2010	13,450	769	5,984	20,203
2011	11,510	648	4,860	17,018

2.2.2.4.1.7 Strakonice – LKST

Status	**Public domestic airport**
Operator	Strakonice Aero Club, phone: 383 321 116
Coordinates	N49°15’06” E013°53’34”
Runway, dimensions and surface	03/21, 900x100 m, grass 13/31, 780x150 m, grass
Frequency	INFO 123,600

This is an uncontrolled airport, providing only an Aerodrome Flight Information Service (AFIS). It has two grass runways with parameters allowing only sports and commercial aircrafts with weight up to 13,000 kg. No significant future change of operating parameters of the airport is taken into account.

As shown in the next table of traffic for 2006 – 2011, the intensity of traffic differs slightly in the individual years. The flights of UL aircrafts (maximum take-off weight up to 450 kg) also represents a significant part of the overall traffic on this airport.

Tab. 37 Frequency of movements at the Strakonice airport

Year	GA movements	GA international	ULL movements	Total
2006	6,166	0	2,302	8,468
2007	5,278	0	5,636	10,914
2008	6,338	78	3,684	10,100
2009	6,896	112	3,076	10,084
2010	3,930	180	2,652	6,762
2011	2,980	165	2,015	5,160

2.2.2.4.1.8 Strunkovice LKSR

Status	**Public domestic airport**
Operator	Prachatice Aero Club, phone: 388 327 124
Coordinates	N49°04’57” E014°04’33”
Runway, dimensions and surface	15/33, 900x23 m, grass
Frequency	INFO 123,500

In addition to the aircrafts owned by the Aero Club (which operates one tug and five gliders), several private aircrafts are operated here, such as Cessna T207 used for parachute drops. A significant part of the traffic includes flights of UL aircrafts owned by private persons.

This is an uncontrolled airport with VFR DAY traffic; the runway allows operations of aircrafts with maximum take-off weight up to 6,500 kg, gliders and UL aircrafts. The airport provides the Aerodrome Flight Information Service (AFIS) and no radio-navigation facilities are installed here. Their future installation and changes in operating parameters of the airport are not taken into account.

Tab. 38 Traffic at the Strunkovice Airport

Year	**Traffic estimate**
2006	4,500
2010	3,500
2011	3,400

The number of movements of GA aircrafts (tug, five gliders plus private Cessna) of this traffic is less than half; so more than half includes operations of UL aircrafts owned by private persons.

2.2.2.4.1.9 Soběslav – LKSO

Status	**Public domestic airport**
Operator	Soběslav Aero Club, phone: 381 521 004
Coordinates	N49°14’48” E014°42’49”
Runway, dimensions and surface	18/36, 740x30 m, grass
Frequency	INFO 122,200

The airport provides the Aerodrome Flight Information Service (AFIS); no radio-navigation facilities are installed here and their future installation is not taken into account.

The main types of operations at this airport are the flights of gliders (starts with aero-tow) and the flights of UL aircrafts.

Tab. 39 Traffic at the Soběslav – LKSO Airport

Year	**GA aircrafts**	**UL aircrafts**	**Total**
2006	1,678	458	2,136
2007	2,080	1,913	3,995

Year	GA aircrafts	UL aircrafts	Total
2008	1,432	1,371	2,073
2009	1,674	1,486	3,160
2010	1,750	1,100	2,850
2011	1,690	1,000	2,700

2.2.2.4.1.10 Tábor – LKTA

Status	**Public domestic airport**
Operator	Tábor Aero Club, phone: 381 263 264
Coordinates	N49°23’28” E014°42’30”
Runway, dimensions and surface	12/30, 1100x130 m, grass 16/34, 850x100 m, grass
Frequency	INFO 122,600

The airport is uncontrolled and provides the Aerodrome Flight Information Service (AFIS). There are neither radio-navigation facilities, nor equipment (runway lighting) for night VFR flights at the airport. Two grass runways can be used for the operations of aircrafts with maximum take-off weight up to 13,000 kg, gliders and UL aircrafts.

Tab. 40 Traffic at the Tábor Airport

Year	Aircraft	GLD	ULL	Total
2006	2,632	1,230	2,038	2011,9
2007	1,860	1,046	2,130	2012,036
2008	2,516	1,180	2,058	2013,754
2009	2,180	1,050	1,980	2014,21
2010	1,220	965	1,840	2978,06
2011	1,260	930	1,860	5,800

2.2.2.4.1.11 Jindřichův Hradec – LKJH

Status	**Public domestic airport**
Operator	J. Hradec Aero Club, phone: 384 321 009
Coordinates	N49°09’03” E014°58’18”
Runway, dimensions and surface	07L/25R, 700x22 m, asphalt 07R/25L, 760x54 m, grass
Frequency	INFO 123,600

The Jindřichův Hradec Airport is situated beyond the border of the considered 40 km zone of the vicinity of the Temelín NPP, at a distance of 44 km east of the Temelín NPP, on the northeast edge of the city of Jindřichův Hradec. Since the airport (like

Hosín and České Budějovice) is provided with an asphalt runway, it is quite often used for flights from abroad.

The airport is uncontrolled and provides the Aerodrome Flight Information Service (AFIS). The airport is intended for operations of VFR DAY; parachute drops are made at the airport in addition to the operations of aircrafts. Runways can handle aircrafts with take-off weight up to 6,500 kg.

Data concerning traffic was estimated at maximum 8,000 movements per year.

2.2.2.4.1.12 Heliports

Overview of heliports being assessed:

- České Budějovice – premises of Jihočeské papírny plant, operator of the České Budějovice Hospital,
- Tábor – hospital, operator of the Municipal Hospital with Outpatient Clinic,
- Jindřichův Hradec, roof of the District Hospital building,
- Temelín – bus parking lot, operator ČEZ, a. s.

České Budějovice, Tábor and Jindřichův Hradec hospital heliports are used for the flights of the Air Rescue Service (LZS) of the Emergency Medical Service of the South Bohemian Region (ZZSJCK). This service covers the area of the whole South Bohemian Region and the Air Rescue Service position is currently at the Hosín Airport.

The Air Rescue Service provides two types of interventions. Primary interventions shall mean a helicopter intervention directly in the field (transport of an injured person from the place of accident to the hospital), when one of the helicopter landings is in the field as required and as needed by attending physicians, and the other landing is then at the heliport of any of the hospitals in the area. Secondary interventions shall mean interventions, during which a helicopter transports a patient (injured person) between the individual hospitals (to specialized workplaces). In this case, the helicopter lands at the heliport of the hospital in question.

The numbers of interventions by the Air Rescue Service in 2004 – 2011 are indicated in the following Tab. 41, taken from the Annual Report of the Emergency Medical Service of the South Bohemian Region [L. 250]. These are the total numbers of interventions on the territory of the South Bohemian Region; primary interventions represent approximately 68% of flights and the remaining percentage covers secondary interventions.

Tab. 41 Number of interventions by Air Rescue Service from heliports on the territory of the South Bohemian Region

	2004	2005	2006	2007	2008	2009	2010	2011
number of interventions by Air Rescue Service	339	385	407	581	638	601	459	512

Whereas the above available sources of information provide only the number of summary interventions in the South Bohemian Region without specifying the individual heliports, the double maximum historically recorded summary numbers of interventions (one intervention includes take-off and landing) were used in the evaluation of the criterion for the assessment of risk arising from the operation of the air facility (see Section 2.2.2.4.3.1 hereof). The conservatism of this procedure lies in the fact that, for analytical purposes, the summary value for all three heliports is assigned to each of them individually.

The heliport at the Temelín site is not used on a regular basis. This heliport is located in the prohibited airspace LK P2 TEMELÍN; its position is marked in the layout in

Fig. 10. The minimum distance of the heliport from the reactor building of the new units is 937 m. Any flight to LK P2 is subject to approval by the Civil Aviation Authority of the Czech Republic.

2.2.2.4.2 Identification of Events

2.2.2.4.2.1 Overview of Accidents

The updated overviews of accidents on the territory of the Czech Republic, which are relevant from the point view of probabilistic analysis, serve as a basis for the evaluation of risk of aircraft falling on structures of the nuclear installation.

For the purposes of statistical monitoring, the accidents are divided into three categories. Information on accidents was obtained for the individual categories of aircrafts as follows [L. 34]:

- Military aircrafts (VOJ) Army of the Czech Republic, Allied Headquarters – Inspectorate, VÚ 2802, 711 11 Olomouc,
- General aviation (aircrafts with weight 450 to 5,700 kg) and civil aviation, transport (aircrafts above 5.7 t) - CIV Civil Aviation Authority, or, as appropriate, Air Accidents Investigation Institute (ÚZPLN), Beranových 130, 199 01 Prague 9, Letňany,
- Air recreational vehicles (aircrafts up to 450 kg) - SLZ Light Aircraft Association of the Czech Republic (LAA), Ke Kablu 289, 102 00 Prague 10.

The number of accidents for the period from the split of the former Czech and Slovak Federal Republic in 1993 is taken into account for calculation of probability of an aircraft crash. In the case of the SLZ category, the accidents have been monitored since the establishment of the category in 1996. Until 1995, these accidents are included in the CIV category.

For the purposes of the assessment of the threat to the nuclear installation due to an aircraft crash, air accident shall mean accidents that may, as a consequence, jeopardize the systems, structures and components important to the nuclear safety, either due to mechanical impact or secondary effects (explosion, fire, secondary fragments, induced vibrations).

Therefore, all the recorded air accidents were subjected a selection process and only those accidents that could, as a consequence, pose a threat to the nuclear installation, were included in well-arranged tables.

The overviews of accidents exclude accidents that happened on the ground as well as accidents associated directly with aircraft take-off and landing at the airport (typically aircraft damaged during landing) because this type of accident is evaluated separately with the effect of near airports.

2.2.2.4.2.2 Accidents in the Category of Air Recreational Vehicles (SLZ)

Tab. 42 Overview of air accidents for the category of air recreational vehicles on the territory of the Czech Republic taken into account for risk assessment

Aircraft category	Aircraft type	Year of accident	Code of accident / phase of flight	Aircraft weight [kg]	Impact velocity [km/h]	Kinetic energy [kJ]	Impact angle [deg]
SLZ	Canard type	1997	ML	300	160	296	60°
SLZ	FOX 912	1997	ML	450	80	111	20-45°
SLZ	ULLA Skyboy	1999	ML	320	120	178	20°
SLZ	ULLA Vlašťovka	1999	ML	400	190	557	20°
SLZ	TL 96 Star	2000	ML	520	100	201	>80°
SLZ	Tukan	2001	ML	450	100	174	45°
SLZ	Straton D 8	2001	ML	450	100	174	>80°
SLZ	Kolibřík	2001	LL	450	80	111	25°
SLZ	Straton D 7	2001	ML	300	80	74	>80°
SLZ	Arius	2002	ML	450	80	111	>45°
SLZ	Zephyr solo	2002	ML	300	200	463	60°
SLZ	Echo	2002	ML	450	70	85	45°
SLZ	Eurostar	2003	ML	450	160	444	60°
SLZ	Straton D 8	2003	ML	450	80	111	30°
SLZ	Piper Cub repl.	2003	ML	300	65	49	30°
SLZ	ULLa Kosák 3	2004	ML	450	80	111	20°
SLZ	P2002 Sierra	2005	ML	560	115	286	<10°
SLZ	ULLa, M7 Ornis	2006	ML	450	140	340	60°
SLZ	KP2.Sova Rapid 200	2006	ML	500	130	326	75°
SLZ	ULLa „Mája“	2006	ML	450	100	174	75°
SLZ	UL Zenair-701	2007	ML	520	>200	802	>60°
SLZ	UL Swing	2007	ML	>530	>150	460	>80°
SLZ	UL EV97	2008	ML	>512	200	790	10 – 20°
SLZ	UL Qualt 201	2008	ML	>480	Approx. 190	669	>80°
SLZ	UL Atec 321	2008	LL	>550	>180	688	>60°
SLZ	UL Sirius	2008	ML	465	>150	404	>60°
SLZ	UL D44BK Fascination	2008	ML	520	>95	181	20°
SLZ	UL Allegro 2000	2008	ML	475	95	165	55°
SLZ	UL Cora	2009	ML	450	Approx. 150	391	>75°
SLZ	ULLa Tecnam P92 Echo	2011	ML	472	170	526	15°

Key to code of accident or phase of flight:

- ML off-airport
- L airport

2.2.2.4.2.3 Accidents in the Category of Civil Aviation (CIV)

Tab. 43 Overview of air accidents for the category of civil aviation on the territory of the Czech Republic taken into account for risk assessment

Aircraft category	Aircraft type	Year of acciden t	Code of accident / phase of flight	Aircraft weight [kg]	Impact velocity [km/h]	Kinetic energy [kJ]	Impact angle [deg]
CIV	R-22	1994	T	650	100	251	70
CIV	Z-226MS	1995	T	1,200	200	1,852	70
CIV	Z-143L	1995	T	1,200	200	1,852	70
CIV	C-172P	1996	T	1,200	100	463	45
CIV	R-22	1996	T	650	100	251	10
CIV	Z-526F	1997	T	1,200	200	1,852	70
CIV	R-22	1997	T	650	100	251	20
CIV	PZL-Kania	1997	T	3,550	200	5,478	20
CIV	Z-126	1999	T	1,200	200	1,852	10
CIV	Z-142	1999	T	1,200	200	1,852	45
CIV	Z-142	1999	T	1,200	200	1,852	10
CIV	Z-142	2000	T	1,200	150	1,042	70
CIV	DR-25	2000	T	1,500	100	579	60
CIV	L-13	2001	T	500	180	625	10
CIV	TB-20	2002	T	1,400	250	3,376	60
CIV	Z-142	2003	T	1,200	150	1,042	45
CIV	L-23	2003	T	510	250	1,230	80
CIV	Z-142	2003	T	1,200	180	1,500	10
CIV	Z142	2007	ML	1090	175	1287	Approx. 45°
CIV	L16 + ASK21	2007	ML	2x 570	Approx. 100, 150	715	>60°
CIV	Zlín Z143	2009	ML	1090	120	606	25-30°
CIV	Cesna 150M	2010	T	750	170	938	0°

Key to code of accident or phase of flight:

- T field
- ML off-airport (for SLZ)

2.2.2.4.2.4 Accidents in the Category of Military Aviation (VOJ)

Tab. 44 Overview of air accidents for the category of military aviation on the territory of the Czech Republic taken into account for risk assessment

Aircraft category	Aircraft type	Year of accident	Code of accident / phase of flight	Aircraft weight [kg]	Impact velocity [km/h]	Kinetic energy [kJ]	Impact angle [deg]	Place of impact
VOJ	Su-22	1993	LÚ	15,000	700	283,565	70	Pístov, JI
VOJ	Mig-21	1996	LÚ	7,500	250	18,084	10	Vlčkovice, RK
VOJ	Su-22	1996	LÚ	15,000	500	144,676	45	Tasovice, BK
VOJ	Mig-23	1996	LÚ	12,000	300	41,667	20	Herálec, ZR
VOJ	L-39	1998	LÚ	5,000	400	30,864	20	Benátky, HK
VOJ	Mig-23	1998	LÚ	12,000	300	41,667	20	Svratka, ZR
VOJ	2x Mig-21	1998	PM	7,000	700	132,330	50	České Vrbné, ČB
VOJ	Mi-24	1998	PM	10,000	330	42,014	45	Slatinice, PV
VOJ	2x Mig-21	1999	N	7,000	700	132,330	80	Macourov, HB
VOJ	Su-22	2000	PM	15,000	350	70,891	5	3 km VPD Náměšť
VOJ	L-29	2000	LÚ	3,000	300	10,417	45	Pohled, CR
VOJ	Mig-21	2000	N	7,000	700	132,330	10	Chotěboř, HB
VOJ	Mig-21	2000	N	7,000	700	132,330	5	Chotěboř, HB
VOJ	L-39	2001	LÚ	5,000	400	30,864	25	Pelhřimov, PE
VOJ	Mi-8S	2001	PM	10,000	100	3,858	30	Okrouhlá, PI
VOJ	L-159	2003	LÚ	6,800	460	55,512	12	VVP Jince

Note: No accidents happened in 2004 to 2009 and 2011. Two air accidents of L39 aircrafts happened in 2010; both accidents happened due to the same cause. The first of the accidents happened in Holice near Pardubice and the second one happened near Biskupice, district of Třebíč. These two accidents are not included in the overview of accidents because their development does not pose a threat to the nuclear installations. During these accidents, the aircrafts were controllable all the time and pilots checked their impact outside the residential and industrial areas.

Key to code of accident or phase of flight:

- ML off-airport (for SLZ)
- T field (for CIV)
- LÚ flight mission
- PM landing manoeuvre (but with impact off the airport)
- N return to airport

2.2.2.4.2.5 Accidents in the Category of Large Airliners

The number of accidents on the territory of the Czech Republic (including the former Czechoslovak Republic) is not sufficient for further statistical evaluation of the probability of accident in different phases of flights and it is insufficient in terms of the assessment of hazard to a ground facility, see Tab. 45. It is obvious that the conditions for the operations of civil air transport have changed during the 1970s and the accident rate cannot be evaluated.

Tab. 45 Airliner accidents on the territory of the Czech Republic

Date	Aircraft	Operator	Point
02/01/1961	Avia-14	ČSA	Prague Hostivice
10/10/1962	Avia-14	ČSA	Brno
24/11/1966	Il-18	Bulgaria	Bratislava
11/10/1968	Avia-14	ČSA	Praha
19/02/1973	Tu-154	Aeroflot	Praha
30/10/1975	DC-9	YAT	Suchdol u Prahy
26/01/1976	DC-9	YAT	Srbská Kamenice
28/07/1976	IL-18	ČSA	Bratislava

The data used for risk analysis of civil aircraft accidents included the analysis of information on airliner accidents worldwide, causes of these accidents and long-term trends in the development of accident rate. The source of information was the publicly available information database [L. 271] and [L. 273].

The probability of an airliner crash onto the nuclear installation ranges from 10^{-9} to 10^{-10}. With regard to the rules of this probability, the risk of airliner crash due to an accident above the territory of the Czech Republic can be neglected.

2.2.2.4.3 Evaluation of Events

2.2.2.4.3.1 Hazards Due to Operation at Near Airports

In accordance with the IAEA recommendation NS-G-3.1 [L. 9], note 11 on page 24, the potential hazards arising from the operation at the aircraft can be neglected if no airports are located in the immediate vicinity of the nuclear installation site and farther airports are used in the extent not exceeding the specified criteria. Criteria and conclusions on air traffic assessment in accordance with such criteria are presented in the next overview.

Airports Located within 10 km from Temelín NPP

There is no airport within this distance from the Temelín NPP site.

A heliport is located in the prohibited airspace LK P2 Temelín (depicted in Fig. 9); this heliport is irregularly used for the activities associated with the operation of the power plant (for location on the premises of the power plant see Fig. 10). These flights can be considered as so-called off-airport operations and in accordance with the standard practice, the hazards associated with flights of helicopters to this heliport can be neglected provided that there are no flights over the structures important from the point of view of nuclear safety. (see page 45 of the standard DOE-STD-3014-96 [L. 245]).

Airports, Situated within 10 to 16 km from Temelín NPP, with Operations Exceeding 500 x D^2 (where D is the distance between the airport and the nuclear installation)

There are neither civil, nor operated military airports within this distance.

Airports, Situated at Distances More than 16 km from Temelín NPP with Traffic Exceeding 1000 x D^2

Within the distances of more than 16 km, there are airports, an overview of which and criteria evaluation are presented in Tab. 46.

Tab. 46 Overview of airports within distances of more than 16 km from ETE3,4 and criteria evaluation for SDV

Airport name	Number of movements/year	Distance from the centre of the nuclear installation at Temelín NPP D [km]	Criterion 1000 . D^2	
České Budějovice Budějovice	6,156	26	676,000	Fulfilled
Hosín	17,018	17.5	306,250	Fulfilled
Soběslav	2,700	26	676,000	Fulfilled
Strakonice	5,160	36	1,296,000	Fulfilled
Strunkovice	3,400	26	676,000	Fulfilled
Tábor	5,800	34	1,156,000	Fulfilled
Jindřichův Hradec	8,000	44	1,936,000	Fulfilled
České Budějovice Budějov. - hospital	1,276	26	676,000	Fulfilled
Tábor - hospital	1,276	34	1,156,000	Fulfilled
J.Hradec - hospital	1,276	43	1,849,000	Fulfilled

The number of movements at the listed airports does not exceed the recommended criteria and hazards arising from the operation at near airports can be neglected. The partial probability $P_{2,IAEA}$ shall equal zero for all considered categories of aircrafts.

2.2.2.4.3.2 Hazards Due to Operation of Nearby Military Air Corridors and Military Training Areas

There are no military installations or training areas, type of firing or bombing practice ranges that may jeopardize the safe operation of the designed structure, located within 30 km from the vicinity of the Temelín NPP [L. 251]. The operation in air corridors can be further neglected if it is situated at a distance of more than 4 km. There are neither air corridors, nor airport take-off and approach corridors running at this distance from the Temelín NPP. Therefore, hazards resulting from this type of air traffic can be neglected. The position of the Temelín NPP in relation to air corridors is depicted in high- and low-altitude en-route charts in Fig. 5 and Fig. 6. The partial probability $P_{3,IAEA}$ shall equal zero for all considered categories of aircrafts.

The total annual probability P_{IAEA} is indicated in Tab. 47.

Tab. 47 Determination of annual summary probability of aircraft crash onto ETE3,4 [accidents/year] for the VOJ and CIV categories

Input data				Probability according to the guide NS-G-3.1 for A_{eff}= 10,000 m^2			
Aircraft category	**Number of accidents**	**Data for the T period [years]**	**A_{eff} [m^2]**	**$P_{1, cat}$**	**$P_{2, cat}$**	**$P_{3, cat}$**	**P_{IAEA}**
SLZ	30	16	0.01	2.378E-07			2.3775E-07
CIV	23	19	0.01	1.535E-07			1.535E-07
VOJ	16	19	0.01	1.068E-07			1.0678E-07
Total SLZ+CIV							3.9125E-07
Total LZ+CIV+VOJ							**4.9803E-07**

The probability of an aircraft crash represents an approximate estimate for an effective intervention area of the facility of 10,000 m^2. The effective intervention area is a structure outline projection on a plain perpendicular to the trajectory of an aircraft crash. Whereas the specific type of the nuclear unit is unknown, the conservative area is specified in accordance with the recommendation defined in Section 5 of the IAEA recommendation NS-G-3.1 [L. 9]. The probability of occurrence of an aircraft crash under the specified conditions reaches the value of $4.98.10^{-7}$. This means that the load by aircraft crash should be taken into account in the design of the nuclear installation as a design event within the solution of external events.

2.2.2.4.4 Events with Non-Neglectable Effects

2.2.2.4.4.1 Aircrafts[14]

The final probability of occurrence of an aircraft crash is determined as a sum of the partial probabilities (accident due to air traffic above the territory of the state, impact of traffic at near airports, traffic in air corridors and training areas). The limit probability is exceeded in all the considered categories of aircrafts, specifically for accidents due to air traffic above the territory of the state. The required resistance of

[14] Aircrafts are airplanes with fixed airfoils.

structures results from the effects of design aircraft crash; only an expert estimate of parameters of the design aircraft is possible without the knowledge of the effective intervention area arising from the particular design. The parameters of the design aircraft shall be determined to make it possible to evaluate the primary effects on the structures of facilities (global and local mechanical effects of impact) as well as the secondary effects (fires or explosions of aviation fuel, flying fragments, vibrations induced by impact).

2.2.2.4.4.2 Helicopters

To use the heliport in the LK-P2 area (in the area of the bus station at the Temelín NPP) without being a source of events jeopardizing the Temelín NPP, helicopter arrivals and departures shall be organized to avoid flights over structures important in terms of nuclear safety [L. 245].

2.2.2.5 LONG-DISTANCE GAS PIPELINE

2.2.2.5.1 Identification of Event Sources

A corridor with four high-pressure gas pipelines supplying natural gas of Russian origin runs along the northwest edge of the premises prepared for the construction of ETE3,4. This involves

- DN 800 PN 75 high-pressure long-distance gas pipeline
- DN 1000 PN 75 high-pressure long-distance gas pipeline
- DN 1400 PN 75 high-pressure long-distance gas pipeline
- DN 200 PN 40 high-pressure gas pipeline Zvěrkovice-Zliv

Furthermore, the Temelín NPP DN 500 PN 4 medium-pressure gas pipeline is in the area, which serves as a gas connection for the Temelín NPP running from Zvěrkovice.

The areas of site facilities and guarded parts of the premises of ETE1,2 and planned guarded areas of ETE3,4 encroach upon the security zone of high-pressure gas pipelines defined in Section 69 of and Annex to Act No. 458/200 Coll. [L. 217], see the layout in Annex to this Initial Safety Analysis Report in Dwg. 3.

To improve operational safety of pipes of long-distance gas pipelines (DN 800 PN 75, DN 1000 PN 75) existing already at the time of deciding on siting of the Temelín Nuclear Power Plant, additional measures were implemented before startup of Temelín NPP Unit 1. These measures included additional visual checks of pipes, repeated checks of X-ray images of their welds, application of additional insulation and installation of conduits; their overview is presented in Chapters 1 and 2 of the report cited in the list of documents as [L. 43].

The DN 1400 PN 75 gas pipeline was designed and completed already during the construction of ETE1,2. Therefore, measures to maximize the safety level highly above the standard limit, required by standard, were created already in its design. The X70 material with minimum guaranteed yield point of 480 MPa was used. The whole section of the line running in parallel with the construction site of the Temelín NPP, starting 100 m before crossing with the road Křtěnov - Temelín at length of 2,232 m to the west, was made with wall thickness of 1,420 x 25.1 mm, which is significantly higher compared to the basic thickness, i.e. 1,420 x 15.6 mm. The summary safety coefficient of this pipe section related to rated yield point and maximum operating pressure is 2.25; the safety level derived from routine and actually supplied yield point approaches 2.5. Special attention was given to corrosion protection. The function of cathodic protection is subjected to periodic in-service inspections [L. 43].

The DN 200 PN 40 gas pipeline Zvěrkovice – Zliv was designed in 1998 to 2003 and constructed in 2004. The potential impact of any accident of this pipeline on the Temelín NPP was solved already during its design and implementation - material selection and piping design helped to reach the safety coefficient of 5.1 (calculated from rated yield point against maximum operating pressure). To stop the development of potential manufacturing cracks in material and remove internal stress in piping, a stress test was performed after the installation had been completed [L. 43].

In addition, all gas pipelines are equipped with automatic safety devices, which in the case of an accident, shut off the flow of gas into the failed section, thus reducing the amount of gas leaked sharing the interaction of the gas pipeline and the environment.

The parameters of natural gas transported by pipeline correspond to its source - Siberian mine fields. This natural gas contains approximately 98.3 to 98.5% of CH_4, about 0.5% of C_2H_6, approximately 0.2% of higher hydrocarbons, about 0.07% of CO_2 and approximately 0.85% of N_2. It is significantly lighter than air, the mean density under normal conditions ranges about 0.69 kg/m^3, i.e. approximately 0.57 of air density. The calorific value under standard conditions ranges at about 37 MJ/m^3, the caloric power is approximately 33.5 MJ/m^3. The alternative natural gas of Norwegian origin taken into consideration is gas of the same substitution group H, thus its properties are basically identical. Due to relatively higher content of higher hydrocarbons, its calorific value and caloric power are higher by several percent (Chapter 4 in the report [L. 43]).

2.2.2.5.2 Identification of Events

Events relating to gas leaked from gas pipeline are divided into two groups by the extent of leak:

- Small leaks insufficient to disintegrate the overburden of gas pipeline and create a crater in the field at the place of leakage (these are leaks with the total area of pipe tightness failure up to approximately 10 mm^2),
- Large leaks in the amount sufficient to erode the overburden and create a crater in the overburden of gas pipeline at the place of leakage (hole in pipe or pipe break).

Significantly lower gas density compared to air is critical for the behaviour of gas leaked from gas pipeline (for numerical values see Section 2.2.2.5.1 hereof). Natural gas disperses in the atmosphere without the possibility of creating a gas cloud moving horizontally in the ground layer of the atmosphere, which could pose a danger of fire to structures or suffocating effects to persons distant from the place of gas leak. In the ground, gas creates a cone-shaped gas body penetrating through the pores in the ground with the volume concentration of gas reducing sideways. On the surface of the ground, gas escapes into the atmosphere. If its concentration above the ground exceeds the lower explosion limit, an accidental initiation can cause its ignition. If no ignition occurs, gas keeps leaking until the defect is detected during walkdown inspection of the gas pipeline. However, if the gas leaked hits an impermeable horizontal layer when penetrating from a gas pipeline leak through the soil upwards (e.g. hard impermeable surfaces, frozen surface layer of soil), it diffuses horizontally through the soil until it finds free access into the atmosphere. If the gas leaked gets into any structure in the vicinity of the source of leak, a gas-air mixture containing natural gas with the concentration in the range between the upper and lower explosion limit can be formed therein.

An overview of events with potential impact on ETE3,4 is shown in Tab. 48.

Tab. 48 Events associated with gas pipeline

Source identification	Risk source name	Event type
EM8.1	small pipe leaks	horizontally diffusing leaks due to explosion of a flammable gas
EM8.2	pipe leak or break	gas leaked into the atmosphere without fire
		gas pipeline fire
		shock wave
		missiles thrown due to leaking gas

2.2.2.5.3 Evaluation of Events

2.2.2.5.3.1 Horizontally Diffusing Leaks

An explosion of natural gas-air mixture in enclosed civil structures occurs on ignition of a mixture formed at the concentration of natural gas and air in the range from approximately 5 to 15%. This ratio shows that energy 1.7 – 5.1 MJ/m^3 is released in the course of the explosion. Such an amount of energy released within approximately 0.01 sec damages, by shock wave, civil structures, process equipment and causes danger to persons present in the place of explosion. Therefore, horizontal diffusing leaks of natural gas cannot be neglected from the point of view of their effects [L. 202].

The frequency of interaction for diffusing leaks is derived from the frequency of detected leaks of gas pipelines in the European Union, which is 5.10^{-6} [1/km, year]. To calculate the frequency of an occurrence of gas pipeline leak resulting in diffusing leak, a gas pipeline section 7.4 km long between the shutoff valves with emergency automatics was selected as a potential source. These automatics automatically close both ends of the section where gas pressure dropped. The shutoff valves on all the segments passing by NPP Temelín are additionally fitted with a special monitoring system, which makes it possible to immediately detect a gas leak from the pipeline even through very small holes.

Frequency of an occurrence of leak is $4 \times 7.4 \times 5 \times 10^{-6} = 1.48 \times 10^{-4}$ [1/year].

The horizontal diffusion of the gas leaked can occur in the case of frozen surface of the ground, supposing for the period of two months in a year, i.e. with a frequency 0.17.

If we assume that the occurrence of gas pipeline leak is independent of the season of the year, the frequency of an occurrence of leak in the period with the possibility of gas diffusion into the premises of the Temelín NPP is

$4 \times 7.4 \times 0.17 \times 5 \times 10^{-6} = 2.5 \times 10^{-5}$ [1/year] (product rounded to 1 decimal place).

The frequency was calculated under conservative assumptions (operational reliability of pipeline in the vicinity of the Temelín NPP was increased above standard design in the European Union; every gas leak in the period of frozen surface of the ground does not result in gas migration into the structures important to the nuclear safety). The assessment of frequency of the possibility of an occurrence of horizontal diffusion shows that this event cannot be neglected with regard to frequency of potential interaction of gas pipeline leak and the Temelín NPP[15].

2.2.2.5.3.2 Gas Leaked into the Atmosphere without Fire

Natural gas leaking from a large gas pipeline leak (hole in pipe or pipe break of gas pipeline) is diluted in the atmosphere and creates a gas-air mixture with characteristic development of natural gas concentration, see Fig. 11. The isoconcentration line of the lower explosion limit of steam-gas mixture may be expected at a height of approximately 36 times the diameter of the pipe [L. 43]. This reduces the probability of an occurrence of fire initiated by the impact of thrown stones.

In the case that gas leaks from a gas pipeline into the atmosphere without its ignition, it is necessary to find out if no unfavourable conditions can occur, during which natural gas could pose a danger to the nuclear installation, its operation or personnel. The possibility of hazards to ETE3,4 due to potential ignition or explosion of a gas cloud (para. 3.48, 3.49, 3.51 of the IAEA standard NS-R-3 [L. 6]) or its suffocating effects (para. 3.48, 3.49, 3.51 of the IAEA standard NS-R-3 [L. 6]) should be assessed.

Whereas natural gas density corresponds to 57% - 59% of air density, depending on temperature conditions, natural gas always moves upwards, reducing its concentration by being diluted with air. This eliminates the formation of a natural gas cloud, which could be transported to the areas of the power plant where it could be the source of fire risk or could fill the attended areas with irrespirable gas.

[15] The follow-up measure – diffusion barrier – is the subject of Sections 2.2.5.4 and 2.2.8.

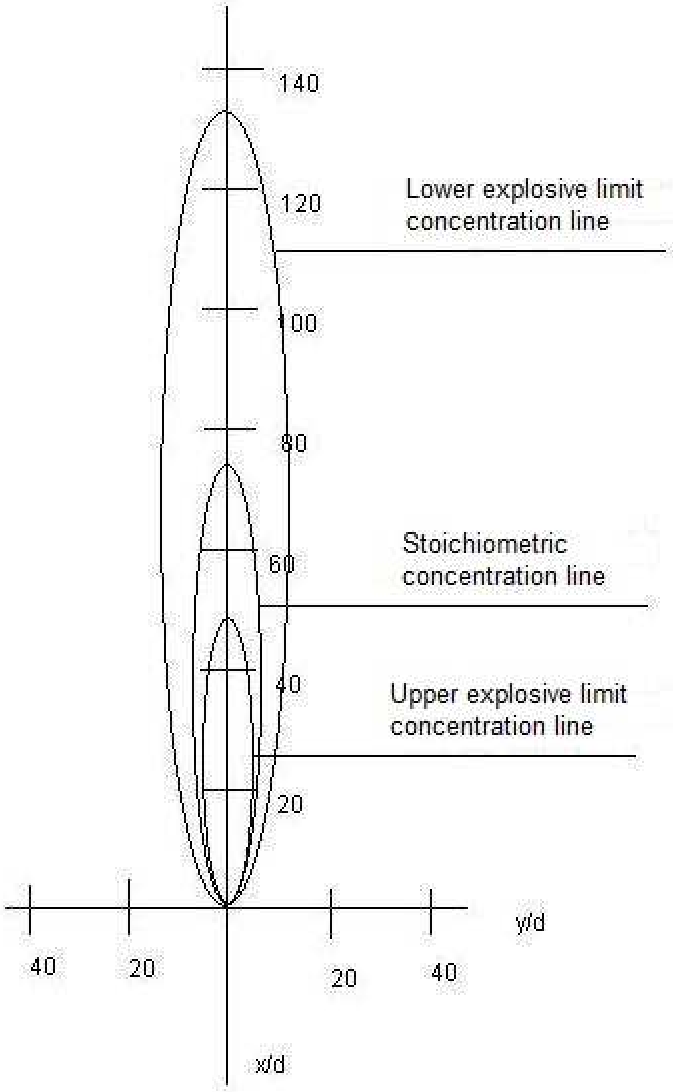

Fig. 11 Gas concentration in gas-air mixture formed by leakage of natural gas

2.2.2.5.3.3 Gas Pipeline Fire

To assess the effect of a hypothetical fire of gas leaked from high-pressure gas pipelines running at the site, a flame was modelled as a thermal radiator in the shape of a cylinder with temperature corresponding to burning temperature of natural gas affecting its environment by emission of heat. Cylinder height and diameter are based on flame dimensions during actual fires of gas pipelines and on theoretical calculations. The analysis documented in Chapter 7 of the report [L. 202] shows that the values of flow of the emitted heat of such modelled fire of gas pipeline beyond the boundary of the 200 m security zone (see Section 69 of and Annex to Act No. 458/2000 Coll. [L. 217] and statement of the owner of the long-distance gas pipeline [L. 216]) will not exceed 4.7 kW/m^2. Not even a long-term action of such flow of the heat affects civil structures. In the case of gas pipelines in the area of the Temelín NPP, the time of potential fire is limited to approximately 5 – 10 min by means of shutoff valves on pipes, which would, in case of pipe integrity failure, reduce the length of pipe from which gas will burn out. For structures located beyond the boundary of the security zone, this defined fire of gas pipeline can be neglected in terms of significance of the effects of interaction of gas pipeline fire and ETE3,4.

However, persons present on the outer edge of the security zone would have to quickly (within 1-2 minutes) find places protected against heat emitting from fire.

In addition, fire of gas pipelines in the section parallel to the fencing of ETE3,4 modelled in the form of two flames 30-40 m away from each other, which would enlarge the surface of thermal radiator, thus increasing the value of the flow of heat emitted on structures in the vicinity, was analysed. However, this conservative fire modelling method would, in real conditions of gas pipelines, correspond to the creation of a large hole or break of two gas pipelines at once (within a maximum of single units of minutes). Whereas the above described failures of gas pipeline are two independent events, the probability of such double failure is less than 10^{-10}/year [L. 202]. In terms of interaction frequency, the described fire can be neglected.

Gas pipeline fire with one flame can be neglected from the point of view of its interaction with ETE3,4. Fire of gas pipelines with two flames on two mutually displaced sections could occur with a frequency insignificant to assure nuclear safety.

2.2.2.5.3.4 Shock Wave

The assessment of the effect of shock wave formed by actual pipe break, i.e. sudden local release of energy accumulated in compressed gas, was based on similar experiment-verified calculation (cited in [L. 44]) of air-borne spread of shock wave during break of an overhead pipe containing natural gas. The following values of overpressure in the front of a shock wave can be expected:

DN 200 below initial pressure of 40 bar 0.1 MPa at an approximate distance of 1.5 m

0.01 MPa at an approximate distance of 15 m

DN 1400 below initial pressure of 75 bar 0.1 MPa at an approximate distance of 5 m

0.01 MPa at an approximate distance of 30 m

Overpressure in the front of shock wave 0.01 MPa corresponds to the value recommended to be considered as design overpressure for safety-related civil structures (shock wave load with overpressure in the front of wave 0.03 MPa was considered in the design for the existing ETE1,2). The value of overpressure in the front of wave 0.01 MPa is determined for the distance from the point of break approximately 30 m, overpressure drops quickly with distance. Therefore, shock wave caused by potential break of high-pressure gas pipeline or in case overhead piping does not pose a danger to structures of the power plant, more than 150 m away. An underground gas pipeline reduces this danger considerably because the shock wave formed would be significantly reduced by action of soil. Therefore, the effect of shock wave can be neglected.

2.2.2.5.3.5 Missiles Thrown due to Leaking Gas

In case of large leak or pipe break of high-pressure gas pipeline, kinetic energy of leaking gas is sufficient to erode the soil surrounding the gas pipeline and scatter it into a broad area. There are registered cases of stones or frozen soil with a weight of several kg being flung to a distance of tens of metres. However, structures of ETE3,4 to be located at a distance more than 100 m cannot be jeopardized.

2.2.2.5.3.6 Summary of Evaluation of Events

The detailed evaluation of risks shows that horizontally diffusing leaks cannot be neglected from the analysed events due to the effects of interaction and frequency of an occurrence of interaction.

Tab. 49 Results of analysis of risks associated with long-distance gas pipeline

Source identification	Risk source name	Event type	Interaction effect	Interaction frequency
EM8.1	small pipe leaks	horizontally diffusing leaks due to explosion of a flammable gas	cannot be neglected	cannot be neglected
EM8.2	pipe leak or break	gas leaked into the atmosphere without fire	can be neglected	-
		gas pipeline fire	can be neglected	-
		shock wave	can be neglected	-
		missiles thrown due to leaking gas	can be neglected	-

Based on the analysis of the effects of events associated with long-distance gas pipeline passing in the vicinity of the Temelín NPP, all events can be neglected as irrelevant, except for small leaks through horizontally diffusing soil. Measures to prevent leaks from diffusing into the areas of the power plant are implemented along the circumference of the premises of ETE1,2. Technically identical design shall be implemented for the premises of ETE3,4 [L. 202].

2.2.2.6 ELECTROMAGNETIC INTERFERENCE

2.2.2.6.1 Identification of Event Sources

Sources of electromagnetic interference can be divided into natural and artificial ones. Detailed information is provided in the report [L. 47].

These sources participate to a different extent in the electromagnetic environment in the site under review. Energy generated by EMI sources can be either emitted to the ambient environment and amplified subsequently by a potentially sensitive device or tapped out to a sensitive device through a power supply line, signal line or other connecting conductors and cables. In general, it is necessary to consider both of the methods of emission from EMI sources, which means emission and propagation along power supply lines. For the ETE3,4 site, the following sources of electromagnetic interference are considered.

Natural Sources of Interference

- Extra-terrestrial sources of cosmic radiation (e.g. supernovas) and the Sun. During solar storms, the magnetic field of the Earth is changed and interference currents are induced in electric networks
- electrostatic discharge (ESD), which is unexpected transfer of charge between bodies with different charges
- Atmospheric discharge (lightning). Interfering effects occur both in the case of direct stroke of lightning at the lightning protective system (LPS), and in the case of indirect strike of lightning in the proximity of outdoor or underground power supply line (see Section 2.4.2.2.8 hereof)

Artificial Sources of Interference

- Sources of transient interference. The main cause thereof are switching processes in the electric network both inside the power plant and in the external electric network (operational switching of consumers, generators, power supply lines, transformers, condensers, switching off failures associated with burning and interruption of electric arc, commutation of electronic inverters, activity of surge voltage protectors, etc.)
- Sources of continuous (sustained) interference. They refer to sources of higher harmonic frequencies (non-linear sources and loads, such as rectifiers, frequency inverters, lighting fixtures). In the area of higher frequencies, continuous interference in the system will be induced by absorption of the emitted interference even from remote systems (e.g. transmitters)

It is therefore obvious that the sources of interference are both local and very remote.

To determine the parameters of the environment in the site of the planned ETE3,4 construction, new measurements were conducted in 03/2011. A network with 15 measuring points, where the monitoring of electromagnetic field was conducted, was established. The monitoring of electromagnetic field covered the following frequency bands: 50 Hz to 10 kHz and 10 kHz to 7 GHz. The measurements concern the level of interference impact generated by the power supply lines, radio and television transmitters, covering transmitters of mobile operators and Wi-Fi devices. The results of the measurements are presented in [L. 47]. The measurements revealed that the intensity of external electromagnetic fields reaches very low values and from the point of view of new equipment installation represents an insignificant source of electromagnetic interference.

Therefore, the decisive source of interference for the newly installed ETE3,4 equipment will be interference generated by the newly installed equipment as such. In addition, there will be interference from natural sources (atmospheric, cosmic, electrostatic) and interference generated by the existing units ETE1,2, the external 400 kV and 110 kV electric network, etc.

The analyses of properties and the design of the necessary consolidation of the external 400 kV and 110 kV electric networks, by means of which the ETE3,4 units will be connected to the electricity network (together with the existing ETE1,2 units), were the subject of the network studies elaborated in 2008 – 2009, see the following reports [L. 233], [L. 234], [L. 235] [L. 236]. The studies cover both the sustained voltage and power situations, and requirements imposed on equipment dimensioning from the point of view of failure conditions and dynamic stability of transmission.

In the vicinity of the ETE3,4 construction site, there are no extensive power or other (transport, ...) direct-current installations. Thus, the risk of stray ground currents is negligible in this location.

2.2.2.6.2 Identification of Events

The aforementioned sources of electromagnetic interference may cause electromagnetic interference (EMI), which is impairment of the equipment/system function. The set of potential event is very complex; it depends not only on the source and nature and intensity of interference, but also on the properties and resistance of the system subject to interference. The events can be both local, and global ones from the point of view of ETE3,4.

The events will be identified in the follow-up stage of the design solution.

2.2.2.6.3 Evaluation of Events

Consequences of the interference impact on equipment and systems can be included in various levels of functional criticality. Depending on the nature and function of the respective system or equipment and the EMI extent, consequences of the events may be either very serious (in terms of safety, economy), or absolutely insignificant.

Consequences of the events depend on the following:

- Nature, source and intensity of interference,
- Links, by means of which interference propagates in the electromagnetic environment,
- Function, properties and resistance of the system and equipment subject to interference.

Electromagnetic interference is quite likely and its occurrence is quite frequent. Some types of interference are present permanently. The specific design solution of ETE3,4 shall cover the protection of the systems, structures and components against interference from identified sources of interference by means of the application of the codes and standards concerning electromagnetic compatibility (EMC).

2.2.3 REQUIREMENTS AND CRITERIA

Requirements and criteria are shared among all the assessed external and internal influences and their sources. Their summary is provided in Tab. 50.

Tab. 50 Requirements and criteria for external and internal human influences

ID	Article	Criteria requirement in Decree No. 215/1997 Coll.	par.	Requirements in IAEA NS-R-3
3.1	Article 5j)	Occurrence of forest areas in the sites selected for siting, where a possible forest fire could endanger the installation or worksite or their operations		

ID	Article	Criteria requirement in Decree No. 215/1997 Coll.	par.	Requirements in IAEA NS-R-3
		or employees		
3.2	Article 5k)	Occurrence of industrial production, energy sources, railroad and water transportation lines and storage of dangerous substances in the close vicinity[16], which could under certain circumstances endanger the installation or worksite or their operations or employees	3.48 3.49 3.50 3.51 Part 1	Activities in the region that involve the handling, processing, transport and storage of chemicals having a potential for explosions or for the production of gas clouds capable of deflagration or detonation shall be identified. Hazards associated with chemical explosions shall be expressed in terms of overpressure and toxicity (if applicable), with account taken of the effect of distance. A site shall be considered unsuitable if such activities take place in its vicinity and there are no practicable solutions available. The region shall be investigated for installations (including installations within the site boundary) in which flammable, explosive, asphyxiant, toxic, corrosive or radioactive materials are stored, processed, transported and otherwise dealt with that, if released under normal or accident conditions, could jeopardize the safety of the installation. This investigation shall also include installations that may give rise to missiles of any type that could affect the safety of the nuclear installation. The potential effects of electromagnetic interference, eddy currents in the ground and the clogging of air or water inlets by debris shall also be evaluated. If the effects of such phenomena and occurrences would produce an unacceptable hazard and if no practicable solution is available, the site shall be deemed unsuitable.

[16] The term "close vicinity" used above represents the territory with a distance of up to 3 km from the boundary of land proposed for the siting (Article2 par. a) of Decree No. 215/1997 Coll. [L. 1]

ID	Article	Criteria requirement in Decree No. 215/1997 Coll.	par.	Requirements in IAEA NS-R-3
3.3	Article 5l)	Intersections between the routes and protection zones of gas, oil and product pipelines and underground stockpiles of goods transported to the plots selected for placement,		
3.4	Article 5m)	Occurrence of buildings of radio and TV broadcasters and their protection zones on plots for placement,		
3.5	Article 5n)	Intersections with protection zones of airports[17], especially their take-off and landing zones and buildings with ground-based aerial equipment, in the zones of sites specified in		
3.6	Article 5q)	Possibility of airplane crash with an impact exceeding the durability of the building containing the installation or worksite, with a probability exceeding 10^{-7} [year $^{-1}$].	3.44 3.45 3.46 3.47	**Aircraft crashes** The potential for aircraft crashes on the site shall be assessed with account taken, to the extent practicable, of characteristics of future air traffic and aircraft. If the assessment shows that there is a potential for an aircraft crash on the site that could affect the safety of the installation, then an assessment of the hazards shall be made. The hazards associated with an aircraft crash to be considered shall include impact, fire and explosions. If the assessment indicates that the hazards are unacceptable and if no practicable solutions are available, then the site shall be deemed unsuitable.
3.7			3.51 Part 2	The region shall be investigated for installations (including installations within the site boundary) in which flammable, explosive, asphyxiant, toxic, corrosive or radioactive materials are

[17] Article 37 of Act No. 49/1997 Coll., on civil aviation and on amendment to Act No. 455/1991 Coll., on commercial activities (Trade Licensing Act)

ID	Article	Criteria requirement in Decree No. 215/1997 Coll.	par.	Requirements in IAEA NS-R-3
				stored, processed, transported and otherwise dealt with that, if released under normal or accident conditions, could jeopardize the safety of the installation. This investigation shall also include installations that may give rise to missiles of any type that could affect the safety of the nuclear installation. The potential effects of electromagnetic interference, eddy currents in the ground and the clogging of air or water inlets by debris shall also be evaluated. If the effects of such phenomena and occurrences would produce an unacceptable hazard and if no practicable solution is available, the site shall be deemed unsuitable.
3.8			3.51 Part 3	The region shall be investigated for installations (including installations within the site boundary) in which flammable, explosive, asphyxiant, toxic, corrosive or radioactive materials are stored, processed, transported and otherwise dealt with that, if released under normal or accident conditions, could jeopardize the safety of the installation. This investigation shall also include installations that may give rise to missiles of any type that could affect the safety of the nuclear installation. The potential effects of electromagnetic interference, eddy currents in the ground and the clogging of air or water inlets by debris shall also be evaluated. If the effects of such phenomena and occurrences would produce an unacceptable hazard and if no practicable solution is available, the site shall be deemed unsuitable.

2.2.4 DOCUMENTS PROVIDING A BASIS FOR THE ASSESSMENT

2.2.4.1 INDUSTRIAL, TRANSPORTATION AND MILITARY BUILDINGS

- IAEA TECDOC-1341 Extreme External Events in the Design and Assessment of Nuclear Power Plants, Vienna, 2010 [L. 15]
- Ing. Miloš Ferjenčík, PhD: Analysis of man-made influences for sections 2.3 and 2.4 - Documents for ISAR ETE3,4, 11/2012 [L. 35]
- Documents for processing Section 2.1 Collection of information and division of external event sources, UJV Řež, a.s. - ENERGOPROJEKT Division Prague, 12/2010 [L. 31]

2.2.4.2 AIR TRAFFIC

- Mgr. Petr Sojka: Air traffic with emphasis on Temelín nuclear power plant security, 2007 [L. 265]
- Mgr. Petr Sojka: Update of air traffic data in the area surrounding the Temelín nuclear power plant - 2010, 12/2010 [L. 34]
- AIP - Aeronautical Information Publication - http://lis.rlp.cz/, Aeronautical Information Service, Air Traffic Control of the Czech Republic (National integrated centre for air traffic control, Navigační 787, 252 61 Jeneč) [L. 271]
- IAEA NS-G-1.5 External Events Excluding Earthquakes in the Design of Nuclear Power Plants, Safety Guide, Vienna, 2003 [L. 7]
- IAEA NS-G-3.1 External Human Induced Events in Site Evaluation for Nuclear Power Plants, Safety Guide, Vienna, 2002 [L. 9]
- DOE STANDARD - DOE-STD-3014-96, Accident Analyses for Aircraft Crash into Hazardous Facilities, U. S. Department of Energy, October 1996 [L. 245]
- Air Traffic information for military aircraft, Joint Forces Headquarters - Němec Miroslav, letter of 05/2012 [L. 251]
- Mgr. Petr Sojka: Update of air traffic data in the area surrounding the Temelín nuclear power plant and overview of airplane accidents, update 04-2012, 04/2012 [L. 249]
- Medical Rescue Service of the South Bohemia Region, annual report 2011 http://www.zzsjck.cz/uploads/pdf/ZZSJcK_-_vyrocni_zprava_2011.pdf [L. 250]
- Methodology for assessment of the threat to the nuclear installation due to an airplane crash and approach to the solution of accidents above the scope of this project including intentional attack with an aircraft, UJV Řež, a.s. - ENERGOPROJEKT Division Prague, 12/2009 [L. 229]
- Publicly accessible accident database of civil aircraft: www.ntsb.gov; www.planecrashinfo.com; www.aviation-safety.net [L. 273]

The selection of accidents in the tables in Section 2.2.2.4 was made based on documents provided by organizations entitled to investigate the causes of aerial accidents and after consultations with specialists in these organizations.

- INSTITUTE FOR SPECIALIZED INVESTIGATION OF AIRPLANE ACCIDENTS, Beranových 130 199 01 Praha 9, Letňany, http://www.uzpln.cz
- Light Aircraft Association of the Czech Republic, Ke Kablu 289, 102 00 Prague 10, HTTP://WWW.LAACR.CZ
- JOINT FORCES HEADQUARTERS - INSPECTORATE, VÚ 2802, 711 11 Olomouc

2.2.4.3 LONG-DISTANCE GAS PIPELINE

- Assessment of influence of gas leaked from the long-distance gas pipeline into the atmosphere on ETE3,4, ÚJV Řež a.s., ENERGOPROJEKT Division Prague, 05/2012 [L. 202]
- Study of the influence of the long-distance gas pipeline and other gas pipelines on the Temelín nuclear power plant, Technical Report, CEPS, a.s., July 2007 [L. 43]
- Assessment of the influence of maximal project accident of the long-distance gas pipeline on NNI buildings, technical report, UJV Řež, a.s. - ENERGOPROJEKT Division Prague, 10/2007 [L. 44]

2.2.4.4 ELECTROMAGNETIC INTERFERENCE

- IAEA NS-G-1.5 External Events Excluding Earthquakes in the Design of Nuclear Power Plants, Safety Guide. Vienna, 2003 [L. 7]
- IAEA NS-G-3.1 External Human Induced Events in Site Evaluation for Nuclear Power Plants, Safety Guide, Vienna, 2002 [L. 9]
- Transmission system codex, revision 11, January 2011 (CEPS a.s.) [L. 230]
- CSN IEC 1000-1-1:95 Electromagnetic compatibility (EMC) – Part 1: General, Part 1: Use and interpretation of basic definitions and terms [L. 231]
- Designation of EMI/EMC environment specifications at the planned ETE3,4 construction site, ABEGU a.s., 04/2011 [L. 47]
- Electromagnetic environment of the Temelín nuclear power plant, Construction premises S1 and S2 for the construction of the 3rd and 4th Block. Test protocol No. P/11/01/19, ABEGY a.s. Laboratory, 03/2011 [L. 232]
- Assessment of PS development variants for the safe distribution of Temelín NNI and EDU. Inspection of dynamic and static stability, CEPS a.s., 31 March 2009 [L. 233]
- Proposal and assessment of PS development variants for the distribution of output from the expanded Dukovany and Temelín nuclear plants with new blocks, EGU Brno, 03/2009 [L. 234]
- Study of short-circuit and voltage ratios in the electrical surroundings of Temelín NPP and NNI EDU, UJV Řež a.s., ENERGOPROJEKT Division Prague, 03/2009 [L. 235]
- Analysis of the possibility of reducing single-phase short-circuit currents for NNI NPP, Assessment of impact on functionality and configurability of the electrical security measures of the Temelín NPP block, UJV Řež a.s., ENERGOPROJEKT Division Prague, 10/2009 [L. 236]

2.2.5 METHODS APPLIED TO THE EVALUATION

2.2.5.1 Assessment System Concepts

The analysis of external and internal human-caused influences on ETE3,4 is based on the process depicted in the figure on

Fig. 12, which reproduces the process scheme specified in recommendation IAEA NS-G-3.1 [L. 9], page 18, on "Fig. 1 General flow diagram for the screening and evaluation procedure".

The assessment of risks based on external events which are caused by industrial, military and ground-based transportation buildings and risks based on internal events (analysis of internal events is the subject of Section 2.3 of this report) uses the same methodology, which is described in this chapter. The default procedure described in the diagram in

Fig. 12 is supplemented for individual groups of external influences by the principles specified in Sections 2.2.5.2 and 2.2.5.5 of this safety report.

The event sources are designated both inside and outside the Temelín NPP premises. The analysis begins by determining potential event sources in the neighbourhood of Temelín NPP. Event sources have the character of buildings and routes where activities with dangerous substances or energies take place. The term "risk source" is used as an equivalent to the standard term "hazard". Hazards in stationary and mobile event sources have the potential to affect the power plant in the form of fires, explosions, creation and spread of flammable clouds, creation and spread of toxic substances, creation and spread of spills of flammable, corrosive or toxic liquids in water, and also (especially in internal sources) creation and spread of clouds of oxidizing substances or asphyxiants.

After a hazard is identified, risk analysis continued by designating and screening possible events. The term "screening" here has the standard meaning, as used in similar English texts.

The screening process has two phases. This removes hazards which do not have a significant effect on the power plant, either based on the distance of the hazard from the plant or the probability of occurrence. Preliminary screening may be carried out either by using the screening distance value (SDV) or by evaluating the probability of the occurrence of the event.

Events which cannot be ruled out by screening are assessed with respect to their frequency and the effects of the interaction of the events with the new nuclear source.

When assessing events, the analysis focuses on interactions of events with ETE3,4 safety-related items. This report assumes that the safety-related items are located in the buildings surrounding the buildings containing the reactors, in the buildings of the diesel-generator stations and in the storage of spent fuel of the new nuclear source.

In compliance with IAEA NS-G-3.1 recommendations [L. 9], the man-made hazards taken into account only include accidental human activities, and non-deliberate actions. Internal events are taken into account for event sources which do not directly participate in the operating states of the blocks of the new nuclear source. The analysis focuses on determining hazards and their possible impacts on ETE3,4. The impacts are assessed based on probabilistic and deterministic criteria. The design bases are defined based on the analysis results (initiating events and parameters). The technical document IAEA TECDOC-1341 [L. 15] was used to analyse extreme and rare events.

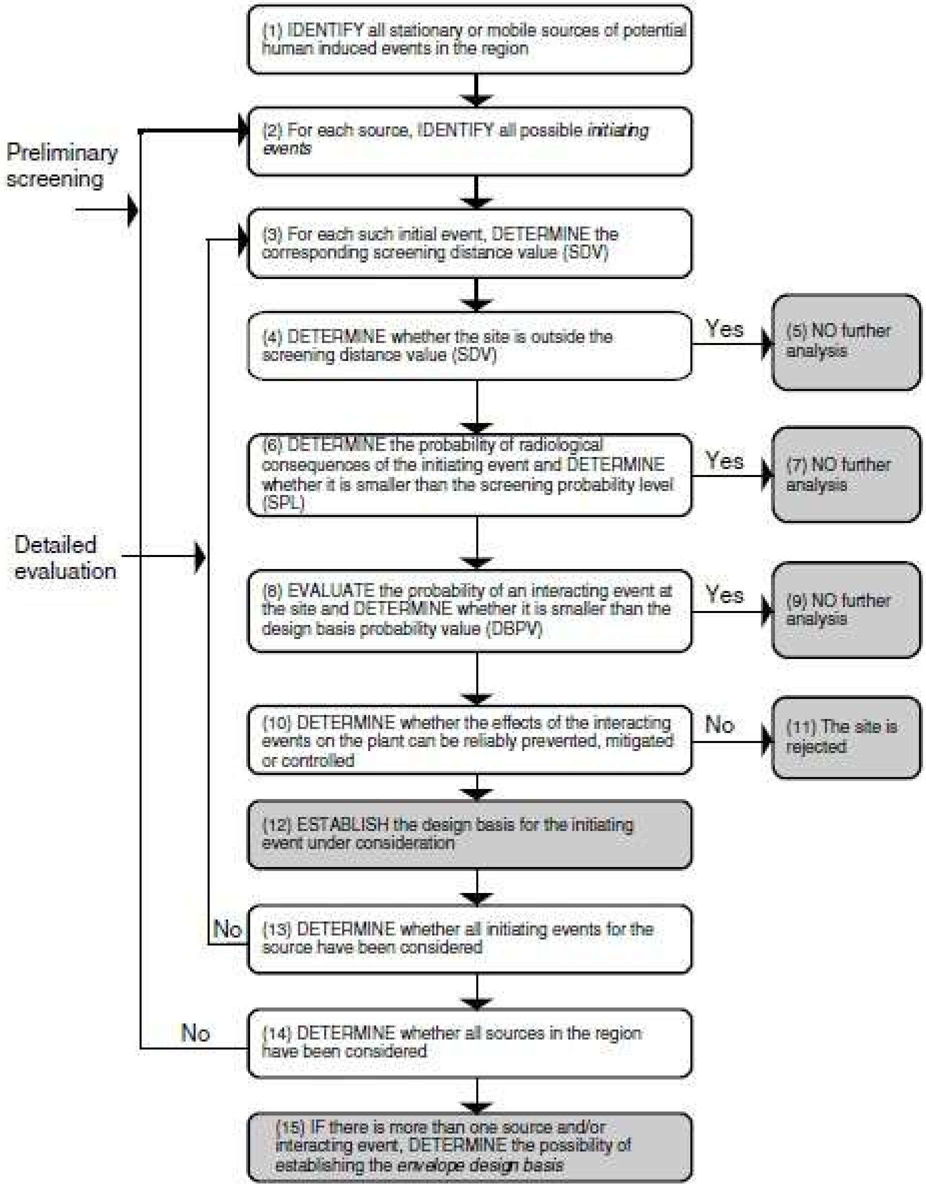

Fig. 12 General flowchart based on [L. 9] for the screening and assessment process (double-framed figures represent finished sequences).

2.2.5.2 INDUSTRIAL, TRANSPORTATION AND MILITARY BUILDINGS

The analysis of risks caused by industrial, transportation and military buildings in the vicinity of ETE3,4 was carried out in compliance with the methodology described in Section 2.2.5.1.

Various events that might interact with the power plant were assumed for each source of risk. In general, the following types of events may be assumed:

- fires
- explosions

- creation and spreading of flammable clouds
- formation and spreading of clouds of toxic substances, formation and spreading of stains of flammable, corrosive or toxic substances in watercourses, as well as formation and spreading of clouds of oxidising substances or asphyxiants.

The assumed types of events are related to the type of hazardous substance in the given source of risk.

Fires are assumed where the hazardous substance is a flammable substance. Explosions are assumed where the hazardous substance is a condensed explosive or taken into consideration where the hazardous substance is a flammable liquid, gas or dispersed dust. The formation and spreading of clouds of flammable substances is associated with fires and explosions and taken into consideration for flammable liquids (however usually not for liquids with hazard level 3) and gases. The formation and spreading of clouds of toxic substances can be the consequence of a fire if the combustion of the flammable substance produces toxic waste gases, or may occur with liquid and gaseous toxic or corrosive substances. The formation and spreading of clouds of oxidising substances are considered where the hazardous substance is an oxidising liquid or gas. Similar considerations also apply for the formation and spreading of clouds of asphyxiants. The potential for occurrence of flying objects threatening the power plant is considered for sources where explosions are conceivable. Flying objects are often called missiles.

Identification of objects and routes outside of the Temelín NPP site is based on the investigation of the surrounding areas, the conclusions of which were summarized in the report [L. 31]. The capacity of buildings with industrial production and the amount of stored dangerous substances was obtained by querying the owners or operators of such buildings. The parameters of the most significant sources were provided by the crisis management body of the Municipal Office of Týn nad Vltavou. The parameters of event sources related to road traffic were provided by the transportation department of the Municipal Office of Týn nad Vltavou.

When identifying these sources, it is possible to replace groups of similar objects by representative objects which have the conservative characteristics of the whole group (smallest distance, maximum amount of dangerous substances, frequency of events is a sum of the whole group, etc.).

A description of the appraisal of the effects and frequencies of interactions of hazard events with the power plant is provided in Section 4.3 of the report [L. 35].

2.2.5.3 AIR TRAFFIC

The methodology for risk analysis of air traffic is based on articles 3.44 to 3.47 of the requirements of the IAEA NS-R-3 standard [L. 6] and on articles 5.1 to 5.20 of the IAEA NS-G-3.1 recommendation [L. 9].

The document used for risk assessment of risks related to aerial transportation was the analysis provided in reports [L. 249], [L. 229] and [L. 262].

The risk assessment of an airplane crashing into Temelín NPP buildings is based on the accident statistics for the territory of the Czech Republic, a summary of which is provided in Sections 2.2.2.4.2.2 to 2.2.2.4.2.5 of this report. Accidents were processed separately for the categories of military aircraft (VOJ), civilian aircraft (CIV) and sports aircraft (aircraft up to 450 kg) (SLZ).

The annual probability $P_{IAEA,cat}$ is calculated as the sum of probability for each of the assessed aircraft categories (VOJ, CIV or SLZ), computed as the sum

$$P_{IAEA,cat} = P_{1,cat} + P_{2,cat} + P_{3,cat} \quad (2.1)$$

$$P_{i,cat} = P_{i,VOJ} + P_{i,CIV} + P_{i,SLL} \qquad \textit{for } i = 1, 2, 3 \quad (2.2)$$

where $P_{1,cat}$, $P_{2,cat}$ and $P_{3,cat}$ represent three independent annual probabilities, representing individual part risks in the given aircraft category:

$P_{1,cat}$ is the annual probability of an airplane crashing on the site due to general air traffic in the given category (VOJ, CIV, SLZ), which was determined based on data on the number of airplane crashes on the monitored territory during a given time period,

$P_{2,cat}$ is the annual probability of an airplane crashing in the given category (VOJ, CIV, SLZ) on the site due to take-off and landing operations of airports in the vicinity,

$P_{3,cat}$ is the annual probability of an airplane in the CIV category crashing on the site due to air traffic on air routes in the vicinity.

The estimated probability $P_{1,cat}$ of an airplane accident affecting the nuclear installation may be determined based on the number of accidents per year per area unit multiplied by the so-called effective area of the building with a risk of damage of safety-related equipment.

The annual probability $P_{1,cat}$ is determined based on the following equation:

$$P_{1,kat} = \frac{n_{kat}}{A_{ref} \cdot T_{kat}} \cdot A_{eff} \qquad \textit{where cat = VOJ, CIV, SLZ} \quad (2.3)$$

where n_{cat} is the number of airplane crashes in the category $_{cat}$ in the period of *T* years,

A_{ref} is the reference area for accident data, specifically the area of the Czech Republic (78,863 km^2),

T_{kat} is the assumed time period for which aerial accident data are available (number of years).

This is the number of years since the dissolution of the Czech and Slovak Federal Republic in 1993.

For SLZ, this is the number of years since the creation of the category in 1996. Until 1995, these accidents were included in the CIV category.

A_{eff} is the so-called effective area of the parts of the site which are at risk of an airplane crash, in km^2

2.2.5.4 LONG-DISTANCE GAS PIPELINE

The consequences of integrity breach of the pipeline and the subsequent escape of natural gas (events EM8.1 and EM8.2) were assessed based on the following events:

- explosive combustion (deflagration) of natural gas caught in closed areas in the power plant
- dispersion of natural gas which escaped to external areas, without occurrence of a fire
- burning of escaped natural gas in external areas
- shockwave created by pipeline breach
- missiles caused by gas leaking from significant pipeline breaches

Out of these assessed hazards following from an integrity breach of the gas pipeline, horizontal diffusion of natural gas is the only hazard which cannot be neglected with respect to the significance of its interaction with ETE3,4.

To solve the risk following from horizontal diffusion of natural gas, an anti-diffusion barrier was included in the Temelín NPP project for this initiating event.

Explosive combustion (deflagration) of natural gas caught in closed areas in the power plant

As was already noted in Section 2.2.2.5.1, natural gas has a lower density than air and thus gas leaked from the pipeline has the potential to diffuse through the surface and spread in the atmosphere. Observed cases included occurrences of the gas diffusing horizontally to surrounding areas due to a frozen part of the surface or impermeable surface caused by e.g. construction work. This allowed the gas to reach even remote areas, where it then escaped to the atmosphere or a closed building, where it created an explosive concentration of gas with a potential of detonation by fire.

The event, based on the model described above for leaks of natural gas from a transit gas pipeline and its diffusion in an enclosed building in the Temelín NPP site, will be referred to as **horizontal diffuse leaks.**

Burning of escaped natural gas in external areas

Based on the analysis listed in the report [L. 44], it follows that escape of gas from the gas pipeline (due to cracks or tears in the pipe) in conditions allowing the transport of gas to the surface and subsequent dispersion will occur after the escaped gas begins to burn in a specific way. This event is further referred to as a **pipeline fire**. Based on the extent of the leak, this could be ground-bound burning of sufficient strength to light grass or, in the hypothetical case of the gas pipeline being ripped apart by a full crosscut, a flame with a thermal output of hundreds of GW. A large-scale leak of gas will create a crater at the gas pipeline, and a flame will be visible above the crater after the gas mixed with air and a corresponding fire initiation. The interaction of a gas pipeline fire with ETE3,4 was assessed based on the analysis of the impact of a hypothetical fire of gas leaking freely from a fully crosscut pipeline.

If the flammable mixture of gas with air is not lit, the gas will disperse into the upper layers of the atmosphere.

Shockwave created by pipeline breach

The assessed event is the impact of a shockwave created by the breach of the pipeline, i.e. a sudden local release of energy accumulated in the pressurized gas.

Due to the fact that project precautions were already taken in ETE1,2, the assessment of the interaction between the gas leak from the long-distance pipeline and ETE3,4 was carried out directly without previous preliminary screening (see the process depicted on Figure

Fig. 12).

2.2.5.5 ELECTROMAGNETIC INTERFERENCE

The EMI and EMC problems are covered by a set of standards (CNS, IEC, EN), which specify the assessment method for the electromagnetic environment with respect to EMI and EMC.

The assessment of the electromagnetic environment was carried out in compliance with these standards within the whole bandwidth of electromagnetic phenomena which is present at the given site. Electromagnetic phenomena were measured as follows:

- measuring of the electromagnetic field at the ETE3,4 construction site in the whole industrial bandwidth of 50Hz to 7GHz and subsequent assessment (the specific method used is listed in the report [L. 47])
- assessment of frequency and intensity of atmospheric discharges on the site based on the local keraunic level (see Section 2.4.2.2 of this report)

Assessment of the properties of the electrification system with respect to its connection to ETE3,4 (with respect to the electromagnetic environment of the industrial frequency) was carried out in compliance with the Transmission System Codex [L. 230]. Analyses the transmission (400 kV) and distribution (110 kV) networks, which will connect both the ETE1,2 and ETE3,4 blocks, are documented in the network studies [L. 233], [L. 234], [L. 235] and [L. 236].

2.2.6 DEFINITION OF THE AREA EXAMINED

In compliance with recommendations of IAEA NS-G-3.1 [L. 9], surface-based hazards are identified up to a distance of 10 km from the Temelín NPP site. In the case of risks which only affect their surroundings in the form of a fire, i.e. not in the form of an explosion or a cloud of toxic substances etc., this distance is reduced to approximately 3 km. The area thus specified complies with all the requirements of the criterion in 3.1.

Identification of airports and their assessment was carried out up to a distance of 40 km from Temelín NPP. Data necessary to assess the risk caused by air traffic was collected for the specified airports. The calculation of the probability of an airplane crashing into a nuclear installation due to general air traffic is based on the ratio of the number of airplane accidents to the reference area, which is the area of the Czech Republic. Airplane accidents are thus monitored across the whole territory of the Czech Republic for all air traffic categories.

With respect to EMI, the assessed area of resources covers the bandwidth range of approximately 50 Hz to 7 GHz. The distances between the source and target of interference cannot be specified. Thus, measurements of the electromagnetic field were carried out to obtain information about the electromagnetic environment at the intended ETE3,4 construction site.

The assessment of the properties of the 400 kV and 110 kV distribution networks is carried out for the electric distribution network of the Czech Republic and its connections abroad.

2.2.7 DETAILED ASSESSMENT OF ALL REQUIREMENTS AND CRITERIA SPECIFIED BY DECREE NO. 215/1997 COLL. IN COMBINATION WITH THE IAEA NS-R-3 STANDARD

2.2.7.1 CRITERION DEFINED BY ARTICLE 5J) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Article 5 par. j) of Decree No. 215/1997 Coll. [L. 1] is reproduced in Tab. 50 under item 3.1.

A possible forest fire (e.g. of the park at the Temelín NPP information centre and the forested part approximately 500 metres from the northwestern part of the fence) was analysed as event ES7.1. Based on Section 6.1.5.4 of the report [L. 35], a detailed analysis was carried out for this event. The conclusion of this analysis was that the interaction between the fire and the power plant may be considered negligible.

A forest fire near the Temelín site will not threaten its devices, workplaces, operation or employees. The intent of siting ETE3,4 at the Temelín site is not in violation of the criteria of Article 5 paragraph j) of Decree No. 215/1997 Coll. [L. 1]

2.2.7.2 CRITERION AND REQUIREMENTS BASED ON ITEM 3.2

2.2.7.2.1 Criterion defined by Article 5 par. k) of Decree No. 215/1997 Coll.

The text of the criterion defined in Article 5 paragraph k) of Decree No. 215/1997 Coll. [L. 1] is reproduced within Item 3.2 in Tab. 50.

At the Temelín NPP site, 7 external stationary event sources were discovered (see Tab. 22) along with 5 external mobile event sources (see Tab. 26 and Tab. 30), which may lead to 41 hazards following from industrial production, energy sources, road, railroad and water transport and storage of dangerous substances. The hazards from these sources were assessed individually in compliance with the methodology described in Section 2.2.5.1. The assessment implies that 3 external mobile hazards may be significant: EM 2.2, EM 2.3 and EM 3.8, which follows from the possible entry of toxic clouds of various substances into the air intake of control rooms. The remaining hazards may be considered negligible, in accordance with the carried out assessment (see Sections 6.2, 6.3 and 6.3 of the report [L. 35]).

The identified hazards EM2.2, EM2.3 and EM3.8 will be taken into account in the selected project solution. The specific project will include the exact and actual placement of the air intake into the nuclear safety-related operating areas, whereas an analysis will document that:

- the hazards are no longer significant, if it shows that toxic clouds cannot reach these, or

- the operating areas significant for nuclear safety will be fitted with technical equipment and procedures to ensure inhabitability if toxic substances are present in the air within the ETE3,4 site.

The intent of siting ETE3,4 in the Temelín site is not in violation with the criterion of Article 5 paragraph k) of Decree No. 215/1997 Coll. [L. 1], since common implementable technical precautions eliminating the identified hazards exist.

2.2.7.2.2 Requirements in Sections 3.48 to 3.51 of IAEA NS-R-3

The text of the requirements in Sections 3.48 to 3.51 of Part 1 of IAEA NS-R-3 [L. 6] is included under item 3.2 in Tab. 77.

The requirements specified in Section 3.48 to 3.51 in compliance with IAEA NS-R-3 [L. 6], with the exception of Section 3.48, comply with the requirements applied to the site through the criteria in Article 5k) of Decree No. 215/1997 Coll. [L. 1], the fulfillment of which was described in Section 2.2.7.2.1. The above-mentioned part, the yet-unprocessed part of the requirements in Section 3.48, includes devices for the transport of gas which may create a gas cloud capable of deflagration or detonation.

To assess the interactions of the long-distance gas pipeline with ETE3,4, the EM 8.1 and EM 8.2 risk analyses were carried out in Section 2.2.2.5. Precautions will be taken for the EM8.1 hazards for ETE3,4, which prevent horizontal diffusion of minor leaks from the long-distance gas pipeline into the Temelín NPP site. Hazards following from EM8.2 including hazards to the nuclear installation by a fire of natural gas leaking from the long-distance pipeline, missiles and the shockwave are insignificant (see Tab. 49).

Additionally, the negotiation of the construction of NPP3, 4 with the owner and operator of the long-distance pipeline [L. 216] imply other safety precautions, which are specified below:

- activities and siting of ETE3,4 buildings at a distance of 200 metres or less from the long-distance gas pipeline will be approved by the gas pipeline administrator
- ETE3,4 buildings important for nuclear safety will not be located in the safety zone around the gas pipeline
- exits from buildings located in the safety zone of the long-distance pipeline will be located on the opposite side from the pipeline

Conclusion: not even a hypothetical accident of the long-distance gas pipeline at the Temelín site would endanger the devices, workplace or operations important for nuclear safety.

2.2.7.3 CRITERION DEFINED BY ARTICLE 5L) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Article 5 paragraph l) of Decree No. 215/1997 Coll. [L. 1] is reproduced in under item 3.3.

A corridor with 3 pipes of the transnational gas pipeline containing pressurised gas is located at a distance of 150 m to the NW from the NPP3, 4 site (see the drawing in Dwg. 3). Article 68 paragraph 2b) of Act No. 458/2000 Coll. [L. 217] specifies a 4 m protective zone for such gas pipelines.

The intent of siting ETE3,4 in the Temelín site is not in violation with the criteria of Article 5 paragraph l) of Decree No. 215/1997 Coll. [L. 1]

2.2.7.4 CRITERION DEFINED BY ARTICLE 5 PAR. M) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Article 5 paragraph m) of Decree No. 215/1997 Coll. [L. 1] is reproduced in Tab. 50 under item 3.4.

Based on data provided by the Czech Telecommunication Office on 8 November 2010 [L. 45], the plot where the Temelín NPP is to be sited does not collide with the safety zones around TV and radio transmitters.

The intent of siting ETE3,4 in the Temelín site is not in violation with the criteria of Article 5 paragraph m) of Decree No. 215/1997 Coll. [L. 1].

2.2.7.5 CRITERION DEFINED BY ARTICLE 5 PAR. N) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Article 5 paragraph n) of Decree No. 215/1997 Coll. [L. 1] is reproduced in Tab. 50 under item 3.5.

The safety zones around airports near the Temelín site (a list of airports is provided in Table Tab. 34, their locations are depicted on Map Fig. 9) do not intersect with the ETE3,4 construction site - see UCL letter [L. 200].

The intent of siting ETE3,4 in the Temelín site is not in violation with the criteria of Article 5 paragraph n) of Decree No. 215/1997 Coll. [L. 1].

2.2.7.6 CRITERION AND REQUIREMENTS BASED ON ITEM 3.6

2.2.7.6.1 Criterion defined by Article 5 par. q) of Decree No. 215/1997 Coll.

The text of the criterion defined in Article 5 paragraph q) of Decree No. 215/1997 Coll. [L. 1] is reproduced within Item 3.6 in Tab. 50.

Based on the analysis of the frequency of accidents of various types of airplanes in the territory of the Czech Republic and the size of the effective area of the ETE3,4 site summarized in Section 2.2.2.4, it follows that an airplane crash of an airplane in the category SLZ, CIV and VOJ which would interact with ETE3,4 can exceed the frequency of 10^{-7} [year^{-1}]. The event of an airplane crash was included in the initiating events and the nuclear safety-related systems, buildings and facilities will be designed and built so that the impact of a reference airplane should not disrupt their operability. The frequency of an airplane crash in other categories, including large commercial aircraft, is significantly below 10^{-7} [year^{-1}].

The intent of siting ETE3,4 at the Temelín site is not in violation of the criterion in Article 5q) of Decree No. 215/1997 Coll. [L. 1], whereas the systems, buildings and devices which are significant for nuclear safety will be designed and built so that the impact of a reference airplane should not disrupt their operability.

2.2.7.6.2 Requirements in Sections 3.44 to 3.47 of IAEA NS-R-3

The text of the requirements in Sections 3.44 to 3.51 of Part 1 of IAEA NS-R-3 [L. 6] is included under item 3.6 in Tab. 50.

The requirements specified in Sections 3.44 to 3.47 of IAEA NS-R-3 [L. 6] correspond to site requirements and an assessment of their characteristics, as specified in the excluding criteria of Article 5 par. q) of Decree No. 215/1997 Coll. [L. 1]. The conclusion to the requirements of Article 5 par. q) of Decree No. 215/1997 Coll. [L. 1] is provided in Section 2.2.7.6.1.

2.2.7.7 REQUIREMENT IN SECTION 3.51 IN PART 2 OF IAEA NS-R-3

The text of the requirement in Section 3.51, Part 2 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 50 under item 3.7.

Measurements [L. 47] have shown that the intensity of external electromagnetic fields reaches very low values and that the new installations do not represent a significant source of electromagnetic interference. The decisive source of interference for the newly installed devices in ETE3,4 will be the interference generated by these newly installed devices themselves as well as interference from natural sources (atmospheric, cosmic, electrostatic interference) and interference generated by ETE1,2, the external 400 and 110 kV distribution networks etc.

There are no extensive energy or other (e.g. transportation) direct current installations near the ETE3,4 site. Thus, the risk of stray ground currents is negligible in this location.

2.2.7.8 REQUIREMENT IN SECTION 3.51 IN PART 3 OF IAEA NS-R-3

The text of the requirement in Section 3.51 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 50 under item 3.8.

Based on the assessment of external impacts caused by human activities and those caused by natural processes, with the exception specified below, no events were identified which could lead to the blockage of the air intake and outlet of the air conditioning systems. Flying objects carried by extreme winds or a tornado cannot simultaneously block the spatially separated systems. The exception mentioned above constitutes the possible impact of snow, which could lead to the simultaneous blockage of air intakes of the air conditioning systems.

The risk of air conditioning intake blockages may be eliminated by implementing precautions in the design of the air conditioning air intake for air conditioning systems important for nuclear safety. The precaution specified above is in compliance with Section 12.4 of IAEA NS-G-1.5 [L. 7].

2.2.8 CONCLUDING ASSESSMENT

Based on the assessment of external influences in compliance with the criteria specified in Article 5 j), k) and q) of Decree No. 215/1997 Coll. [L. 1], and based on the requirements of Sections 3.48 to 3.51 and 3.44 to 3.47 and 3.51, the siting of ETE3,4 on the planned plots is in compliance with the specified criteria and requirements.

The analysis of external events has led to the identification of initiating events [18], which may have a significant impact on ETE3,4 and which cannot be neglected due to low frequency. As a protection against all these events and their effects on ETE3,4, it is possible to implement precautions into the project design which will

[18] The term "initiating event" here corresponds to the standard meaning of the term "design basis external events" as defined in section 1.6 of IAEA NS G-1.5 [L. 7]

protect the constructions, systems and equipment important with respect to nuclear safety and the operating staff.

The project design of ETE3,4 will ensure protection against the following initiating events:

- railroad and road transport of ammonia (EM2.3 and EM3.8), which would create a toxic ammonia cloud in case of a spill
- railroad transport of nitrogen acid (EM2.2), which would create a toxic cloud in case of a spill
- diffusion of gas leaking from the transnational gas pipeline through soil in the horizontal direction under the impermeable ground (EM8.1), whereby the gas could accumulate in the NPP building, explode, and induce fire
- airplane crash needs to be taken into account within the nuclear installation project as an initiating event for systems, structures and components important for nuclear safety.
- potential blockage of the VAC suction openings will be avoided by the design of the VAC equipment of the nuclear safety-related systems.

The ETE3,4 project will comply with the norms and standards of electromagnetic compatibility (the EMC concept). The project will define the electromagnetic environment in which the systems and equipment will work. The systems must be designed so as to be able to work in the environment without deterioration of operability due to EMI. Furthermore, the systems/equipment must meet the limits for interference emitted into the electromagnetic environment. Additional provisions as may be appropriate must address issues of the interface (coupling pathways) between the systems and electromagnetic environment (protections, interference limiters, etc.). The EMC norms and principles are specified in further detail in [L. 47].

The interfaces between the blocks and the network and between the blocks themselves will comply with EMC principles.

To exclude the negative effects of EM2.2, EM2.3 and EM3.8, the outputs of the ETE3,4 project will be equipped to prevent the intake of a toxic cloud by air conditioning systems for the operating areas (the control room).

To exclude the negative influences of the events specified in item c), the design basis of ETE1,2 and ETE3,4 include a gas proof barrier on the exposed parts of the perimeter of the power plant and a detection system for possible gas leaks diffusing towards the NPP.

To exclude the negative influences of the events specified in item d), the design basis of ETE3,4 include the requirement that the nuclear safety-related systems, structures and components will be able to withstand a crash of an item similar to an airplane with the entered specifications. The effective area of nuclear safety-related buildings in ETE3,4, 10,000 m^2 (see note 12 on page 25 of IAEA NS-G-3.1 [L. 9]) corresponds to a small military design airplane weighing 7 tons with an impact speed of 200 m/s. After choosing the specific construction project for ETE3,4 the effective area of these buildings will be provided in further detail; if this area decreases significantly, it would be possible to consider a smaller civil aircraft instead[19].

As described in the network studies referred to in Section 2.2.2.6 , the connection of ETE1,2 to the 400 kV and 110 kV network will be modified and, furthermore, the networks themselves will be modified and reinforced so that, for the NPP blocks, they should:

- ensure evacuation and distribution of the generated power in all states required by the Distribution System Codex
- be stable during disturbances and failures of the network and at the NPP blocks (short-term, medium-term, long-term dynamics)
- possess an adequate short-circuit resistance, ability to limit and dampen spread of failures and rapidly and selectively switch off severe failures
- possess an adequate supplying capacity, reliability and short-circuit strength for operational and standby supply to the blocks

[19] For the existing NPP blocks, the design aircraft considers civil aircraft with a weight of 7 tons and an impact speed of 100 m/s.

2.3 RISK OPERATIONS INSIDE THE POWER PLANT

2.3.1 SCOPE OF THIS SECTION

Section 2.3 follows up on Section 2.2, which presents the results of the analysis of external man-made influences on ETE3,4 and supplements it with the results of the risk analysis caused by sources located in the NPP site (internal sources). The analysis focused on data about hazards located in the ETE1,2 site.

Identification of hazards following from the operation of ETE3,4 will only be available after processing a specific project of the selected type of power plant, and will be included in the pre-operating safety report. Fulfilling the requirements of the documentation will imply the assumption that the operation of ETE3,4 will not be a source of unacceptable risk that would compromise the safety level of the power plant.

2.3.2 SUMMARY OF FACTS

2.3.2.1 IDENTIFICATION OF EVENT SOURCES

Based on the design of ETE1,2 and the excerpt from the database of hazardous substances used in the operation of ETE1,2 [L. 32], stationary event sources were identified within the ETE1,2 site as well as the dangerous substances and types of events that are associated with these sources. A summary is provided in Tab. 51.

Tab. 51 Internal stationary event sources

Marking	building, activity	Hazardous substances
IS1	Chemical storage (building 592/01), refilling and storage	ammonia water, hydrazine hydrate, nitrogen acid, sulphuric acid
IS2	storage of technical gases in cylinders (building 642/01), storage	acetylene, argon-methane compound, oxygen, methane, propane-butane, hydrogen
IS3	hydrogen warehouse management (building 643/01), storage and refilling	hydrogen
IS4	diesel management (building 703/04), storage and refilling	diesel, oil, used oil
IS5	gas boiler room (building 410), transport to boilers	natural gas
IS8	nitrogen warehouse management (building 643/01), storage and refilling	nitrogen
IS10	asphalt and bitumen RAO at AASB storage (building 801), storage and refilling	asphalt, bitumen RAO
IS11	gas fire-extinguishers in the machine room and exchanger station (buildings 490 and 491), storage and refilling	carbon dioxide

Marking	building, activity	Hazardous substances
IS12	oil systems in the primary production block, in the cooling water pumping stations, in the machine room and in the diesel-generator stations (buildings 800, 574, 490, 445 and 442), storage and refilling	oil
IS13	fuel system for diesel-generator stations (buildings 442 and 445), storage and refilling	diesel, oil

Note: The labels IS6, IS7 and IS9 were used for event sources relevant for another variant of the ETE3,4 design. To keep the original analysis of external influences consistent, the numbering of event sources was not changed.

The list of mobile event sources provided in table Tab. 52 is based on the internal ETE1,2 communication system - roads and railroads and hazardous substances transported within the plant. The solution diagram of the transport routes based on the ETE1,2 project is provided in the drawing in Annex Dwg. 4.

Tab. 52 Internal mobile event sources

Marking	route, activity	Hazardous substances
IM1	railroad line to track towards chemical storehouse (from railroad gateway along building 801 up to object 703/01 and through the switch to the south-west, south around building 641/01 to building 592/01), transportation	ammonia water, nitric acid, sulphuric acid
IM2	roadway towards chemical storehouse, (from roadway gateway towards building 592/01), transportation	hydrazine hydrate
IM3	roadway towards the storehouse of technical gases in cylinders, (from roadway gateway between buildings 592/01 and 655/01 and in front of building 590/01 to the left south-east towards building 642/01), transportation	acetylene, argon-methane compound, oxygen, methane, propane-butane, hydrogen
IM4	roadway towards the hydrogen warehouse (behind the roadway gateway to the left between the administration building and building 701/01 and to the south-east up to building 643/01), transportation	hydrogen
IM5	railroad line to track for tapping of diesel oil at diesel oil warehouse (from railroad gateway along building 801 up to building 703/04), transportation	diesel oil
IM6	roadway towards the tapping spot at the diesel oil warehouse, (from roadway gateway between buildings 592/01 and 655/01 and in front of building 641/01 to the left south-east towards building 703/04), transportation	diesel oil

Marking	route, activity	Hazardous substances
IM7	roadway towards the nitrogen warehouse (behind the roadway gateway to the left between the administration building and building 701/01 and to south-east up to building 643/01), transportation	nitrogen

Descriptions of these sources are included in more detail in Section 6.4.2 of the report [L. 35].

2.3.2.2 IDENTIFICATION OF EVENTS

Events listed in tables Tab. 53 (stationary sources) and Tab. 54 (mobile sources) can originate from event sources defined in Section 2.3.2.1.

Tab. 53 Event types considered in internal stationary event sources

Marking	event source	substances	event types
IS1	chemical warehouse	ammonia water, hydrazine hydrate, nitrogen acid, sulphuric acid	spreading of toxic substance clouds
IS2	store of technical gases in cylinders	acetylene, argon-methane compound, methane, hydrogen,	fire, explosion, missiles, creation and spreading of flammable substances
		oxygen	spreading of oxidising substance clouds
		propane-butane	fire, explosion, missiles, creation and spreading of flammable substances
IS3	hydrogen warehouse management	hydrogen	fire, explosion, missiles, creation and spreading of flammable substances
IS4	diesel oil management	diesel, oil, used oil	fire, explosion, missiles, creation and spreading of flammable substances
IS5	gas-burning boiler room	natural gas	fire, explosion, missiles, creation and spreading of flammable substances
IS8	nitrogen warehouse management	nitrogen	spreading asphyxiant clouds, missiles
IS10	asphalt and bitumen RAO at AASB storage	asphalt, bitumen RAO	fire
IS11	gas extinguishing systems of the 2nd block	carbon dioxide	spreading asphyxiant cloud
IS12	oil systems of the 2nd block	oil	fire
IS13	fuel system for diesel-generator stations west of the 2nd block	diesel oil	fire

Tab. 54 Event types considered in internal mobile event sources

Marking	event source	substances	event types
IM1	railway line towards chemical storehouse	ammonia water, nitric acid, sulphuric acid	spreading of toxic substance clouds
IM2	roadway towards chemical storehouse	hydrazine hydrate	spreading of toxic substance clouds
IM3	roadway towards store of technical gases in cylinders	acetylene, argon-methane compound, methane, hydrogen	fire, explosion, missiles, creation and spreading of flammable substances
		oxygen	spreading of oxidising substance clouds
		propane-butane	fire, explosion, missiles, creation and spreading of flammable substances
IM4	roadway towards hydrogen warehouse	hydrogen	fire, explosion, missiles, creation and spreading of flammable substances
IM5	railway line towards diesel oil warehouse	diesel oil	fire, explosion, missiles, creation and spreading of flammable substances
IM6	roadway towards diesel oil warehouse	diesel oil	fire, explosion, missiles, creation and spreading of flammable substances
IM7	roadway towards nitrogen warehouse	nitrogen	spreading of asphyxiant cloud, missiles

An overview of risk sources related to stationary event sources in ETE1,2 is listed in table Tab. 55

Tab. 55 Risk sources in internal stationary event sources

Marking	risk source	description	location
IS1.1	presence and replenishment of sulphuric acid in chemical storehouse	4 × 80 m^3 96% sulphuric acid and 50 t 96% sulphuric acid in railroad cistern, 2/year	building 592/01 and place for pumping chemicals
IS1.2	presence and replenishment of nitric acid in chemical storehouse	2 × 16 m^3 65% nitric acid and 25 t 65% nitric acid in barrels in a truck, 2/year	building 592/01 and place for pumping chemicals
IS1.3	presence and replenishment of hydrazine hydrate in chemical storehouse	2 × 6 m^3 15% hydrazine hydrate and 5 t 15% hydrazine hydrate on a truck, 2/year	building 592/01 and place for pumping chemicals
IS1.4	presence and replenishment of ammonia water in chemical storehouse	2 × 25 m^3 25% ammonia water and 25 t 25% ammonia water in railroad cistern, 2/year	building 592/01 and place for pumping chemicals

Marking	risk source	description	location
IS2.1	presence of acetylene in the store of technical gases in cylinders	acetylene, 10 cylinders per 10 kg	building 642/01
IS2.2	presence of hydrogen in the store of technical gases in cylinders	hydrogen, 2 cylinders per 1 kg	building 642/01
IS2.3	presence of oxygen in the store of technical gases in cylinders	oxygen, 50 cylinders per 50 kg	building 642/01
IS2.4	presence of propane-butane in the store of technical gases in cylinders	propane-butane, 12 cylinders per 10 kg	building 642/01
IS2.5	presence of methane in the store of technical gases in cylinders	methane, 2 cylinders per 50 kg	building 642/01
IS2.6	presence of argon-methane in the store of technical gases in cylinders	argon-methane, 10 cylinders per 50 kg	building 642/01
IS3.1	presence of hydrogen in hydrogen warehouse management	hydrogen, 311 kg of hydrogen in 32 bundles	building 643/01
IS3.2	replenishment of hydrogen in hydrogen warehouse management	hydrogen, up to 311 kg of hydrogen in 32 bundles and 78 kg of hydrogen in 8 bundles on a truck, 52/year	building 643/01 and the area in front of it
IS4.1	presence of diesel oil in diesel oil warehouse management	up to 4 × 1000 m^3 of diesel oil in overflow traps	building 703/04
IS4.2	replenishment of diesel oil in diesel oil warehouse management from railroad cisterns	up to 4 × 1000 m^3 of diesel oil in overflow traps and up to 6 × 50 m^3 of diesel oil in railroad cisterns, 1/year	objects 703/04, shelter and hall for tapping for diesel oil
IS4.3	replenishment of diesel oil in diesel oil warehouse management from road tankers	up to 4 × 1000 m^3 of diesel oil in overflow traps and up to 33 m^3 of diesel oil in road tanker, 9/year	objects 703/04, shelter and hall for tapping for diesel oil
IS5.1	the inlet of natural gas feed line for the boiler room,	natural gas, average 200 mm, pressure 200 kPa	along the south-east wall of building 410
IS8.1	presence of nitrogen in nitrogen warehouse management	3 × 30 m^3 and 3 × 10 m^3 of liquid nitrogen	building 643/01

Marking	risk source	description	location
IS8.2	replenishment of nitrogen in nitrogen warehouse management	nitrogen, road tanker 14 m^3	building 643/01 and the area in front of it
IS10.1	presence of asphalt and bitumen RAO at AASB	asphalt, 2 × 45 m^3, bitumen RAO, 3 × 37,200 l barrels	north-east and north from building 801 AASB
IS11.1	presence of carbon dioxide in gas extinguishing system of the 2nd block	carbon dioxide, 60 × 15 Nm^3	building 491/02
IS12.1	presence of oils in oil systems of the 2nd block	oils, 89 m^3 in total	buildings 490/02, 491/02, 800/04,05,06, 584/02, 445/01,02,03, 442/03
IS13.1	presence of diesel oil and oils in fuel system of diesel-generator stations west of the 2nd block	diesel oil, totally 224 m^3, oils, 20 m^3 in total	buildings 442/03 and 445/01, 02, 03

The risks specified in Tab. 56 are linked to transportation lines (railroad and road) leading through the current Temelín NPP (i.e. ETE1,2 and shared premises of the area).

Tab. 56 Internal risks with source in transportation lines in Temelín NPP

Marking	risk source	description	location
IM1.1	railway transportation of sulphuric acid to the chemical storehouse	50 t 96% sulphuric acid in railroad cistern, 5/year, 2200 m	from railroad gateway to building 592/01
IM1.2	railway transportation of nitric acid to the chemical storehouse	25 t 65% nitric acid in 40 barrels in a truck, 5/year, 2200 m	from railroad gateway to building 592/01
IM1.3	railway transportation of ammonia water to the chemical storehouse	25 t 25% ammonia water in railroad cistern, 5/year, 2200 m	from railroad gateway to building 592/01
IM2.1	road transport of hydrazine hydrate to the chemical storehouse	5 t 15% hydrazine hydrate in a truck, 5/year, 200 m	from roadway gateway to building 592/01
IM3.1	roadway transport of acetylene to store of technical gases in cylinders	acetylene, 5 cylinders per 10 kg, 65/year, 650 m	from roadway gateway to building 642/01
IM3.2	roadway transport of hydrogen to store of technical gases in cylinders	hydrogen, 1 cylinder with 1 kg, 5/year, 650 m	from roadway gateway to building 642/01

Marking	risk source	description	location
IM3.3	roadway transport of oxygen to store of technical gases in cylinders	oxygen, 25 cylinders per 50 kg, 130/year, 650 m	from roadway gateway to building 642/01
IM3.4	roadway transport of propane-butane to store of technical gases in cylinders	propane-butane, 6 cylinders with 10 kg, 10/year, 650 m	from roadway gateway to building 642/01
IM3.5	roadway transport of methane to store of technical gases in cylinders	methane, 1 cylinder with 50 kg, 5/year, 650 m	from roadway gateway to building 642/01
IM3.6	roadway transport of argon-methane to store of technical gases in cylinders	argon-methane, 5 cylinders per 50 kg, 23/year, 650 m	from roadway gateway to building 642/01
IM4.1	roadway transport of hydrogen to hydrogen warehouse	8 cylinder bundles with hydrogen, i.e. approximately 78 kg of hydrogen, 130/year, 500 m	from roadway gateway to building 643/01
IM5.1	railway transport of diesel oil to diesel oil warehouse	6 x 50 m^3 in railroad cisterns, 3/year, 1600 m	from railroad gateway to building 703/04
IM6.1	roadway transport of diesel oil to diesel oil warehouse	33 m^3 in road tankers, 23/year, 700 m	from roadway gateway to building 703/04
IM7.1	roadway transport of nitrogen to nitrogen warehouse	nitrogen, in road tankers 14 m^3, 500 m	from roadway gateway to building 643/01

2.3.2.3 EVALUATION OF EVENTS

As follows from the preliminary evaluation of the effects and frequency of ETE3,4 interactions with events linked to risk sources located in the NPP1.2 site in accordance with methodology described in Section 2.2.5 of this report, a more detailed analysis is required for events and risks listed in tables Tab. 57 and Tab. 58. An evaluation description of individual events linked to internal risk sources is included in Section 6.4.4 of the report [L. 35].

Tab. 57 Risks linked to internal stationary sources requiring a more detailed analysis

source marking	risk source name	event type	interaction effect	interaction frequency
IS1.1	presence and replenishment of sulphuric acid in chemical storehouse	spreading of a toxic cloud	cannot be neglected	cannot be neglected

source marking	risk source name	event type	interaction effect	interaction frequency
IS1.2	presence and replenishment of nitric acid in chemical storehouse	spreading of a toxic cloud	cannot be neglected	cannot be neglected
IS1.3	presence and replenishment of hydrazine hydrate acid in chemical storehouse	spreading of a toxic cloud	cannot be neglected	cannot be neglected
IS1.4	presence and replenishment of ammonia water in chemical storehouse	spreading of a toxic cloud	cannot be neglected	cannot be neglected
IS2.3	presence of oxygen in the store of technical gases in cylinders	spreading of an oxidising cloud	cannot be neglected	cannot be neglected
IS3.1	presence of hydrogen in hydrogen warehouse management	spreading of a flammable cloud	cannot be neglected	cannot be neglected
		explosion of a flammable cloud	cannot be neglected	
IS3.2	replenishment of hydrogen in hydrogen warehouse management	spreading of a flammable cloud	cannot be neglected	cannot be neglected
		explosion of a flammable cloud	cannot be neglected	
IS4.1	presence of diesel oil in diesel oil warehouse management	liquid fire	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
IS4.2	replenishment of diesel oil in diesel oil warehouse management from railroad cisterns	liquid fire	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
IS4.3	replenishment of diesel oil in diesel oil warehouse management from road tankers	liquid fire	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
		explosion of a flammable cloud	can be neglected	

source marking	risk source name	event type	interaction effect	interaction frequency
IS8.1	presence of nitrogen in nitrogen warehouse management	spreading of asphyxiant	cannot be neglected	cannot be neglected
IS8.2	replenishment of nitrogen in nitrogen warehouse management	spreading of asphyxiant	cannot be neglected	cannot be neglected
IS11.1	presence of carbon dioxide in gas extinguishing system of the 2nd block	spreading of asphyxiant	cannot be neglected	cannot be neglected
IS12.1	presence of oils in oil systems of the 2nd block	fire	cannot be neglected	cannot be neglected
IS13.1	presence of diesel oil and oils in fuel system of diesel-generator stations west from the 2nd block	fire	cannot be neglected	cannot be neglected

Tab. 58 Tab. Risks linked to internal mobile sources requiring a more detailed analysis

source marking	risk source name	event type	interaction effect	interaction frequency
IM1.1	railway transportation of sulphuric acid to the chemical storehouse	spreading of a toxic cloud	cannot be neglected	cannot be neglected
IM1.2	railway transportation of nitric acid to the chemical storehouse	spreading of a toxic cloud	cannot be neglected	cannot be neglected
IM1.3	railway transportation of ammonia water to the chemical storehouse	spreading of a toxic cloud	cannot be neglected	cannot be neglected
IM2.1	road transport of hydrazine hydrate to the chemical storehouse	spreading of a toxic cloud	cannot be neglected	cannot be neglected
IM3.3	roadway transport of oxygen to store of technical gases in cylinders	spreading of an oxidising cloud	cannot be neglected	cannot be neglected
IM4.1	roadway transport of hydrogen to hydrogen warehouse	spreading of a flammable cloud	cannot be neglected	cannot be neglected
		explosion of a flammable cloud	cannot be neglected	
IM5.1	railway transport of diesel oil to diesel oil warehouse	liquid fire	cannot be neglected	cannot be neglected

source marking	risk source name	event type	interaction effect	interaction frequency
IM6.1	roadway transport of diesel oil to diesel oil warehouse	liquid fire	can be neglected	cannot be neglected
		spreading of a flammable cloud	cannot be neglected	
IM7.1	roadway transport of nitrogen to nitrogen warehouse	spreading of asphyxiant	cannot be neglected	cannot be neglected

Based on the detailed assessment of the effects and frequencies of interactions described in Section 6.4.5 of the report [L. 35], it follows that, similarly to external risks, it is not possible to neglect the following risks related to the "spreading of toxic cloud" event type.

2.3.2.4 EVENTS WITH NON-NEGLECTABLE EFFECTS

2.3.2.4.1 IS1.2 presence and replenishment of nitric acid in the chemical storehouse

Identification of events

The undesired event begins with the leak of nitric acid. In case of storage, the acid would leak from the storage tank into the 6 x 12 m overflow trap, and in case of pumping, the acid would leak from the wagon into the pumping overflow tank (this also applies for cistern wagons, not only for the usually used barrels). In either case, the acid would not leak into the free environment and it would only be evaporating from a relatively small surface area. Undesirable events are spilling of the acid, its evaporation and the spreading of a cloud of acid.

Creation and spreading of a toxic cloud during storage

The spilling of the acid in the whole surface of the overflow trap, evaporation from the resulting puddle and spreading of toxic vapours are an undesirable event. The overflow trap has a surface area of 72 m^2. The equivalent diameter of the puddle would be approximately 9.6 m. Based on available data, the tension of 65% acid vapours (UN 2031) at 311 K (38°C) does not even reach the value of 25 mm Hg, i.e. approximately 3300 Pa. The calculation in IS1.2-1, as documented in the report [L. 35], shows that at a wind speed of 1 m/s the amount of evaporated acid would be up to 0.01 kg/s. The acid vapours are heavier than air, and so the model for the spread of heavy gases was used. Calculation IS1.2-2 shows that the reach of ERPG-2 concentration at the same wind speed and under the worst stability conditions could be at most 770 m. It was verified that greater wind speeds lead to more favourable results. In either case, this is more than the distance of the closest ETE3,4 buildings important for safety. The impacts of the interaction of the event with ETE3,4 cannot be neglected.

Frequency

The estimation of frequency is based on the estimation of the frequency of defects of stable pressureless tanks, and is thus approximately 2 x 3 x 10^{-5}/year. The event thus cannot be neglected based on its frequency.

Creation and spreading of a toxic cloud during pumping

The spilling of the acid in the whole surface of the overflow trap for pumping, evaporation from the resulting puddle and spreading of toxic vapours are an undesirable event. The overflow trap has a surface area of 210 m^2. The equivalent diameter of the puddle would be approximately 16 m. Based on available data, the tension of 65% acid vapours (UN 2031) at 311 K (38°C) does not even reach the value of 25 mm Hg, i.e. approximately 3300 Pa. The calculation in IS1.2-3 shows that at a wind speed of 1 m/s the amount of evaporated acid would be up to 0.026 kg/s. The acid vapours are heavier than air, and so the model for the spread of heavy gases was used. Calculation IS1.2-4 shows that the reach of ERPG-2 concentration at the same wind speed and under the worst stability conditions could be at most 1400 m. The stability category F used in this calculation however represents the worst possible night-time conditions, and it may be assumed that pumping will only take place during the day. Calculation IS1.2-5 shows that the reach of ERPG-2 concentration at the same wind speed and under the worst daytime stability conditions could be approximately at most 430 m. Again, the air conditioning intake for ETE3,4 could be affected. Thus the effects of the interaction on ETE3,4 cannot be neglected.

Frequency

The pumping event frequency will probably be significantly affected by the possibility of human error. It cannot be assumed that the event may be neglected due to its frequency.

2.3.2.4.2 IS1.4 presence and replenishment of ammonia water in the chemical storehouse

Identification of events

The undesirable event is initiated by a leak of ammonia water. In case of storage, the liquid would leak from the storage tank into the 6 x 12 m overflow trap, and in case of pumping, the liquid would leak from the wagon into the pumping overflow tank (this also applies for cistern wagons, not only for the usually used barrels). In either case, the liquid would not leak into the free environment and it would only be evaporating from a relatively small surface area. Undesirable events are spilling of ammonia water, its evaporation and the spreading of a cloud of ammonia.

Creation and spreading of a toxic cloud during storage

The spilling of the liquid in the whole surface of the overflow trap, evaporation from the resulting puddle and spreading of toxic vapours are an undesirable event. The overflow trap has a surface area of 72 m^2. The equivalent diameter of the puddle would be approximately 9.6 m. Based on data in the safety sheet, the tension of vapours of the liquid at 293 K (20°C) reaches approximately 64000 Pa. Calculation IS1.4-1 shows that with a wind speed of 1 m/s up to 0.087 kg/s would evaporate from the puddle. Calculation IS1.4-2 shows the reach of concentration at the same wind speed and under the worst stability could be approximately at most 520 m. Air intake

of control rooms in ETE3,4 which can be at a distance of approximately 400 m, would be affected already by ERPG-2 concentration.
The impacts of the interaction of the event with ETE3,4 cannot be neglected.

Frequency

The estimation of frequency is based on the estimation of the frequency of defects of stable pressureless tanks, and is thus approximately 2 x 3 x 10^{-5}/year. The event thus cannot be neglected based on its frequency.

Creation and spreading of a toxic cloud during pumping

The spilling of the liquid in the whole surface of the overflow trap for pumping, evaporation from the resulting puddle and spreading of toxic vapours are an undesirable event. The overflow trap has a surface area of 210 m^2. The equivalent diameter of the puddle would be approximately 16 m. Based on data in the safety sheet, the tension of vapours of the liquid at 293 K (20°C) reaches approximately 64000 Pa. Calculation IS1.4-3 shows that with a wind speed of 1 m/s up to 0.23 kg/s would evaporate from the puddle. Calculation IS1.4-4 shows that the reach of ERPG-2 concentration at the same wind speed and under the worst stability conditions could be at most 640 m. The stability category F used in this calculation however represents the worst possible night-time conditions, and it may be assumed that pumping will only take place during the day. Calculation IS1.4-5 shows that the reach of concentration ERPG-2 at the given wind speed and under the worst day-time stability could be approximately at most 190 m. The calculation indicates that air intakes of control rooms in ETE3,4 would not be affected already by ERPG-2 concentration.

The impacts of the interaction of the event with ETE3,4 can be neglected.

2.3.2.4.3 IM1.1 railway transportation of sulphuric acid to the chemical storehouse

Identification of events

Only one event is analysed: spilling of the acid, its evaporation and the spreading of a cloud of acid.

Creation and spreading of a toxic cloud

The extreme event would be the spilling of the whole cistern content, evaporation from the resulting puddle and spreading of toxic vapours. 50 t of the acid has a volume of approximately 27.3 m^3 and could be spilled on the area of approximately 1365 m^2. The equivalent diameter of the puddle would be approximately 42 m. Based on available data, the tension of acid vapours at 293 K does not even reach the value of 1 mm Hg, i.e. approximately 133 Pa. The calculation IM1.1-1 shows that at a wind speed of 1 m/s the amount of evaporated acid would be up to 0.011 kg/s. The acid vapours are heavier than air. Calculation IM1.1-1, where vapours are considered to be a neutrally buoyant gas and the source is modelled as areal, shows that reach of concentration ERPG-2 at the same wind speed and under the worst day-time conditions would be approximately at most 100 m. Calculation IM1.1-3, where the acid vapour is considered a heavy gas and admixing rate of oxygen above the puddle is included in the calculation, shows that the reach of concentration ERPG-2 would not exceed 190 m (see report [L. 35]]

With regards to the fact that the distance of the transport spot to the air intakes for control rooms ETE3,4 could be less then 70 m, the impact of interaction of events with ETE3,4 cannot be neglected.

Frequency

It is possible to evaluate the frequency based on an estimation of line section length along buildings with items important for safety in km, frequency of passages of sets with acid on the line per year and estimation of accident probability during the passage of a single set on 1 km of the line. The section along buildings with air intakes does not exceed 0.4 km. This length must be extended by $2 \times 0.190 \times (1 - (0.070/0.190)^2)^{1/2}$. Evaluation of event frequency is therefore $0.75 \times 5 \times 5\times10^{-7} = 1.875\times10^{-6}$/year. If we consider that the leak creation is estimated for one out of four accidents and the probability of spreading the cloud towards ETE3,4 buildings with items important for safety is approximately 0.38, the frequency can be evaluated as 1.78×10^{-7}/year. Therefore this event cannot be neglected even with regards to the frequency.

2.3.2.4.4 IM1.2 railway transportation of nitric acid to the chemical storehouse

Identification of events

Only one event is analysed: spilling of the acid, its evaporation and the spreading of a cloud of acid.

Creation and spreading of a toxic cloud

The extreme and with regards to the package of acid also very improbable event is the spilling of all acid in the truck, evaporation from the resulting puddle and spreading of toxic vapours. Twenty five tons of 65% acid has a volume of approximately 18.8 m^3 and could be spilled on the area of approximately 940 m^2. The equivalent diameter of the puddle would be approximately 35 m. Based on available data, the tension of 65% acid vapours (UN 2031) at 311 K (38°C) up to 25 mm Hg, i.e. approximately 3300 Pa. The calculation IM1.2-1 documented in the report [L. 35] shows that at a wind speed of 1 m/s the amount of evaporated acid would be up to 0.011 kg/s. The acid vapours are heavier than air. Calculation IM1.2-2,where the acid vapour is considered a heavy gas and admixing rate of oxygen above the puddle is included in the calculation, shows that the reach of estimated concentration ERPG-2 at the same wind speed and under the worst day-time stability could be approximately at most 1100 m. Thus the effects of the interaction on ETE3,4 cannot be neglected.

Frequency

It is possible to evaluate the frequency based on an estimation of line section length along buildings with items important for safety in km, frequency of passages of sets with acid on the line per year and estimation of accident probability during the passage of a single set on 1 km of the line. The section along buildings with air intakes equals the whole length of the line. Evaluation of event frequency is therefore 2.2 × 5 × 5×10-7 = = 5.5×10^{-6}/year. If we consider that the leak creation is estimated for one out of four accidents (which is probably too strict for transportation in barrels) and the probability of spreading the cloud towards buildings ETE3,4 with items important for safety does not exceed at any location approximately 0.38, the frequency can be evaluated as 5.2×10^{-7}/year. Therefore this event cannot be neglected even with regards to the frequency.

2.3.2.4.5 IM1.3 railway transportation of ammonia water to the chemical storehouse

Identification of events

Only one event is analysed: spilling of ammonia water, its evaporation and the spreading of a cloud of ammonia.

Creation and spreading of a toxic cloud

The extreme event would be the spilling of the whole cistern content, evaporation from the resulting puddle and spreading of toxic vapours. Twenty five tons of the liquid would have a volume of approximately 22 m^3 and could be spilled on the area of approximately 1100 m^2. The equivalent diameter of the puddle would be approximately 37 m. Based on data in the safety sheet, the tension of vapours of the liquid at 293 K (20°C) reaches approximately 64000 Pa. Calculation IM1.3-1 shows that at a wind speed of 1 m/s the amount evaporated from the puddle would be 1.11 kg/s at most. Calculation IM1.3-2 (source modelled as areal) carried out in the report [L. 35] shows the reach of concentration at the same wind speed and under the worst day-time stability could be approximately at most 770 m. The effects of the interaction on ETE3,4 cannot be neglected.

Frequency

It is possible to evaluate the frequency based on an estimation of line section length along buildings with items important for safety in km, frequency of passages of sets with the liquid on the line per year and estimation of accident probability during the passage of a single set on 1 km of the line. The section along buildings with air intakes equals the whole length of the line. Evaluation of event frequency is therefore 2.2 × 5 × 5×10-7 = = 5.5×10^{-6}/year. If we consider that the leak creation is estimated for one out of four accidents and the probability of spreading the cloud towards buildings ETE3,4 with items important for safety does not exceed 0.38, the frequency can be evaluated as 5.2×10^{-7}/year. Therefore this event cannot be neglected even with regards to the frequency.

2.3.2.4.6 IM2.1 road transport of hydrazine hydrate to the chemical storehouse.

Identification of events

Only one event is analysed: spilling of the solution, its evaporation and the spreading of a cloud of acid.

Creation and spreading of a toxic cloud

The extreme event would be the spilling of the content of all containers, evaporation from the resulting puddle and spreading of toxic vapours. Five tons of the solution would have a volume of approximately 5 m^3 and could be spilled on the area of up to 500 m^2 (presupposes spilling partially on a hardened surface). The equivalent diameter of the puddle would be approximately 25 m. Based on data in the safety sheet, the tension of vapours of the liquid at 293 K (20°C) reaches approximately 20mbar, i.e. 2000 Pa. Calculation IM2.1-1 documented in the report [L. 35] shows that at a wind speed of 1 m/s the amount evaporated from the puddle would be 0.020

kg/s at most. Calculation IM2.1-2 documented in the report [L. 35], with modelled areal leak source, shows the reach of concentration ERPG-2 at the same wind speed and under the worst day-time stability could be approximately at most 430 m. The effects of the interaction on ETE3,4 cannot be neglected because the distance of ETE3,4 items important for safety could be approximately the same.

Frequency

It is possible to evaluate the frequency based on an estimation of line section length along buildings with items important for safety in km, frequency of passages of vehicles with the liquid on the line per year and estimation of accident probability during the passage of a single vehicle on 1 km of the line. The section along buildings with air intakes equals the whole length of the line, i.e. 0.2 km. Evaluation of event frequency is therefore 0.2 × 5 × 5×10-6 = = 5×10^{-6}/year. If we consider that the probability of spreading the cloud from the area of the internal roadway towards ETE3,4 buildings with items important for safety is approximately 0.25, the frequency can be evaluated as 1.25×10^{-6}/year. Therefore this event cannot be neglected even with regards to the frequency.

2.3.3 REQUIREMENTS AND CRITERIA

Requirements and criteria specified for industrial, transportation and military buildings in Section 2.3.3 for external influences are valid also for internal influences where meaningful.

With regard to the character of risk sources, those requirements and criteria listed in table Tab. 50 shall be applied that are included under item 3.2 following the conditional criterion according to Article 5 par. k) of Decree No. 215/1997 Coll. [L. 1].

2.3.4 DOCUMENTS PROVIDING A BASIS FOR THE ASSESSMENT

- Documents for processing Section 2.1 Collection of information and division of external event sources, UJV Řež, a.s. - ENERGOPROJEKT Division Prague, 12/2010 [L. 31]
- A list of hazardous substances in Temelín NPP is processed for the preparation of the Protocol on classification of the building into a relevant group according to the act on preventing of major accidents, ČEZ, a.s. - Temelín NPP [L. 32]

2.3.5 METHODS APPLIED TO THE EVALUATION

To evaluate the impact of risks linked to sources located in the Temelín NPP area, the methodology described in Section 2.2.5.1 of this report was used without any modifications.

Identification of risk sources inside the Temelín NPP area is based on the identification of hazardous substances. Substances suitable for inclusion among risk sources consist of substances classified as flammable, toxic, corrosive or oxidising that are used in great amounts or in physical states favourable for accidents. Great amounts, with regards to liquids and solid substances, refer to hundreds of kg and more; physical states favourable for accidents refer mainly to gaseous state or the state of over-heated liquid.

The potential for occurrence of missiles threatening ETE3,4 is considered only for risk sources where explosions are conceivable.

2.3.6 DEFINITION OF THE AREA EXAMINED

Section 2.3 includes the assessment of risks linked to sources located inside the ETE1,2 area as delimited by the fence of the guarded area. The assessed area in Section 2.3 follows the area assessed in chapter 2.2 and together they form a complete area that can be considered for placing risk sources that can affect the ETE3,4.

A scheme of the Temelín NPP area with internal risk sources is included in the Annex to this report on a design marked as Dwg. 4.

2.3.7 DETAILED ASSESSMENT OF ALL REQUIREMENTS AND CRITERIA SPECIFIED BY DECREE NO. 215/1997 COLL. IN COMBINATION WITH THE IAEA NS-R-3 STANDARD

2.3.7.1 CRITERION DEFINED BY ARTICLE 5 PAR. K) OF DECREE NO. 215/1997 COLL.

The text of the criterion defined in Article 5 paragraph k) of Decree No. 215/1997 Coll. [L. 1] is reproduced within Item 3.2 in Tab. 50.

Based on risk assessment following from possible internal event in Section 2.3.2 a total of six internal risk sources was identified, whose interaction with ETE3,4 cannot be neglected. They are as follows:

- IS1.2 presence and replenishment of nitric acid in the chemical storehouse
- IS1.4 presence and replenishment of ammonia water in the chemical storehouse
- IM1.1 railway transportation of sulphuric acid to the chemical storehouse
- IM1.2 railway transportation of nitric acid to the chemical storehouse
- IM1.3 railway transportation of ammonia water to the chemical storehouse
- IM2.1 road transport of hydrazine hydrate to the chemical storehouse.

Elimination of internal risks following from the operation of ETE1,2 will be ensured through the realization of corresponding measures in the project ETE3,4; they are technically feasible and therefore do not exclude the placement of ETE3,4 in the proposed locality.

2.3.7.2 REQUIREMENT IN SECTIONS 3.48 TO 3.51 OF IAEA NS-R-3 STANDARD

The text of the requirements in Articles 3.48 to 3.51 of Part 1 of IAEA NS-R-3 standard [L. 6] is included under item 3.2 in Tab. 50.

Requirements in Sections 3.48 to 3.51 of IAEA NS-R-3 standard[L. 6] form part of the methodology used to identify risks for ETE3,4 and their assessment in Section 2.3.2. Results of the risk source assessment were used as a basis for assessment of the criterion in accordance with Article 5 par. k) of Decree No. 215/1997 Coll. [L. 1], included in Section 2.3.7.1

2.3.8 CONCLUDING ASSESSMENT

Based on the assessment of internal influences consisting in the operation of ETE1,2 in compliance with the criteria specified in Article 5 j), k) and q) of Decree No. 215/1997 Coll. [L. 1], and based on the requirements of Sections 3.48 to 3.51 and 3.44 to 3.47 and 3.51, the siting of ETE3,4 on the planned plots is in compliance with the specified criteria and requirements.

The analysis of accidents that can occur in ETE1,2 lead to the identification of initiating events corresponding to spread of toxic clouds, which may have a significant impact on ETE3,4 and which cannot be neglected due to low frequency. To protect against all these events and their impacts on NPP3, 4, it is possible to implement precautions into the project design which will protect the constructions, systems and equipment which is important with respect to nuclear safety and the operating staff. For protection against initiating events following from external influences, project measures implemented for protection against external influences are sufficient (see Section 2.2.8). This consists of the protection against spreading of toxic clouds (including burning emissions). Control rooms and other attended nuclear safety-related areas are equipped with technical equipment and trained procedures preventing the entry of toxic substances. This spreading of toxic clouds will be be included among the initiating events, and resistance against these events has to be an initiating parameter.

2.4 METEOROLOGY

2.4.1 SCOPE OF THIS SECTION

This section processes the meteorology conditions for the ETE3,4 construction site in accordance with IAEA requirements and recommendations, namely with IAEA standard NS-R-3 [L. 6] and IAEA recommendations NS-G-3.4 [L. 12].

Following the processing of these conditions, they were assessed according to relevant criteria, as defined by Decree No. 215/1997 Coll. [L. 1].

Description, analysis and evaluation are provided for:

- extreme and rare meteorological phenomena
- meteorological data required for assessing the dispersion of radioactive substances in the air

2.4.2 SUMMARY OF FACTS

2.4.2.1 SPECIFICATION OF EXTREME VALUES OF METEOROLOGICAL PARAMETERS

2.4.2.1.1 Temperature

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 4.29 – 4.35. A summary of facts is included in table Tab. 59.The table includes recommended estimates of maximum and minimum temperatures, derived from series of annual (seasonal) maximum or minimum required temperature characteristics derived using various methods (see Section 2.4.5.2). Values in the table were obtained by an expert evaluation of estimations, performed separately for each time series of each considered station. Results for each stations are listed in tables available in the Annexes of this report [L. 203]. Columns in the "Repetition time" group include estimations of maximum and minimum air temperatures that can be expected once in a hundred years and once in ten thousand years calculated by various estimation methods. These methods presuppose a stationary time series of the input data. The last column of table Tab. 59 is based on a climate development simulation using scenarios based on last results of climatic change models as of 2060 [L. 212] with their expert extension up to 2080. Values listed as "estimation up to year 2080" are quantities that will be reached or exceeded with probability 10-4 and lower.

For minimum temperatures, the hypothetical possibility, discussed among the professional public, that, in the case of extremely high levels of warming, the reverse trend, i.e. cooling, may occur in 2060, was also considered. However, not even at maximum or minimum temperatures, it is not expected that the limits listed in the second numeric column in Tab. 59 (repetition time 10,000 years). will be reached or exceeded as of 2080

Tab. 59 Recommended estimations of maximum and minimum temperatures

Recommended temperature estimations (°C)	Estimation method	Repetition time		Estimation until 2080
		100 years	1000 years	
Maximum instantaneous temperature	MLEEV	42.0	52.0	below 48
Maximum 6 hour average	MLEEV	38.5	46.2	below 44
Maximum 24 hour average	MLEEV0	31.6	38.8	below 37
Maximum 7 day average	MLEEV	27.7	34.5	below 34
Minimum instantaneous temperature	MLEEV/LRSEV	-35.6	-47.0	-40
Minimum 6 hour average	LRSEV	-32.0	-47.0	-38
Minimum 24 hour average	MLEEV0	-24.3	-37.3	-30
Minimum 7 day average	MLEEV	-20.4	-33.1	-25

Note: Estimations calculated according to [L. 208]. The MLEEV0 method consists of the Gumbel distribution; the MLEEV method is the three-parameter generalized extreme distribution. For more detail see Section 2.4.5.2.

2.4.2.1.2 Wind velocity

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 4.5 – 4.11.

As follows from tables available in the Annexes of the background documentation [L. 203], different estimation methods yield slightly different results, as well as data based on a calendar year and on the season (a year calculated from July to the following June). Requirements of methodological recommendation IAEA No. 50-SG-S11A [L. 17] are best met by the MLEEV0 method (Gumbel distribution). If differences between estimations listed in the table are considered, estimations recommended in the following table appear to be the best estimations.

Tab. 60 Recommended estimation values at 1s and 10 s and 10 min of wind load (m/s)

Repetition time	100 years	10,000 years
Wind gust 1 s (m/s)	48	65
Wind gust 10 s (m/s)	37	50
10-minute mean speed (m/s)	28	38

Note: One-second gusts were measured. 10-second gusts are not monitored; it is a qualified estimation based on partial tests within the Czech Hydrometeorological Institute.

2.4.2.1.3 Precipitations

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 4.12 – 4.21.

As follows from tables available in the Annexes of the Czech Hydrometeorological Institute report [L. 203], different estimation methods yield slightly different results. Requirements of methodological recommendation IAEA No. 50-SG-S11A [L. 17] for 3 to 24-hour intervals are best met by the MLEEV0 method Gumbel distribution); for a 15-minute interval the most suitable method is the estimation method based on the logarithmic-normal distribution percentile of all 15-minute summaries from 2 mm measured in the observation period [L. 209]. If differences between estimations are considered, estimations recommended in the below table Tab. 61 appear to be the best estimations.

Tab. 61 Recommended estimations of storm rains with repetition time of 100 years

Amount/time	Estimation method	Repetition time 100 years
mm/15 min	99. percentile of log-norm. distribution	42.0
mm/3 hours	Gumbel	66.5
mm/6 hours	Gumbel	83.1
mm/24 hours	Gumbel	109.2

The repetition time of 100 years described above is used as standard for storm rains for the purpose of designing building construction and dimensioning of rain sewage. Repetition time of 100 years for storm rains meets the requirements stipulated in Sections 3.9 and 3.10 of IAEA NS-R-3 [L. 6]. In order to enable the comparison of extreme meteorological phenomena values with other meteorological parameters, rain storms estimation in [Tab. 62] are provided for the repetition time of 10,000 years.

Tab. 62 Recommended estimations of storm rains with repetition time of 10,000 years

Amount/duration	Estimation method	Repetition time of 10,000 years
mm/15 min	expert estimation[20]	58.0
mm/3 hours	MLEEV0	116.1
mm/6 hours	MLEEV0	139.7
mm/24 hours	MLEEV0	182.0

2.4.2.1.4 Snow conditions

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 4.22.

The Annexes of the Czech Hydrometeorological Institute report [L. 203] include a table with mean values of the water rate of snow with the probability of exceeding once in one hundred and once in ten thousand years. This also includes weather changes, when the precipitation can be absorbed into the snow. The table includes

[20] It is not possible to make a meaningful formal calculation of the 10,000-year value based on the series of monitored data in the area. Therefore the MLEEV0 estimations for 1 hour and 10,000 years and MLEEV0 for 15 min and 1,000 years were used to create an expert estimation, which also considered the maximum 15 min rain amount obtained by ombrographic monitoring on all stations in the Czech Republic The observed period differs at each station, ranging from 10 to 100 years. Ombrographic (pluviographic) monitoring using an ombrograph (pluviograph) consists of monitoring precipitation with a graphic output and reading accuracy of one minute for daily records and 10 minutes for weekly records. Results of automatic measurements of precipitation intensity were also taken into consideration (minute records, measured in Southern Bohemia since the late nineties (gradually putting stations into operation), whereas the results of all mentioned measurements were confronted and revised by comparison with daily values of a classical rain meter.

results obtained by using three different estimation methods in the Xtremes software [L. 208]. The MLEEV method is three-parameter generalized extreme distribution; the MLEEV0 method consists of Gumbel distribution.

Recommended initiated values for 100 and 10,000 years are stated in the table Tab. 63 that correspond to the MLEEV method for area within 25 km considering the length of data series and larger number of considered stations in this area range. Larger areas are too influenced by the vicinity of Šumava. We would like to stress that the stated initiated values are water rates of the snow cover, i.e. snow cover values already expressed by a water column.

Tab. 63 Recommended estimations of the water rates of snow (water column height in mm) for the repetition time of 100 and 10,000 years.

Parameter	Estimation method	Repetition time	
		100 years	10,000 years
Water rate of snow (mm of water column)	MLEEV	104	180

2.4.2.2 SPECIFICATION OF RARE METEOROLOGICAL PHENOMENA

2.4.2.2.1 Data sources for rare meteorological phenomena

The data were processed in accordance with IAEA NS-G-3.4 [L. 12], article 3.13.

Both data monitored on the Temelín meteorological station and on the nearest climatological stations in Tábor and České Budějovice, as well as data from meteorological station Kocelovice were used to calculate the long-term average number of days with the phenomenon (storms, icing and hailstones) in accordance with the recommendation in [L. 12]. Precipitation measuring station Týn nad Vltavou also provides a long series of monitoring in the surroundings of Temelín. Monitoring from this station provides data on precipitation from 1897 to 1981, with long gaps. However, there are no available data on the incidence of meteorological phenomena. Location of stations is listed in tab. 5_1 and indicated on maps 5_3 and 5_3 in the Annexes of this report [L. 203]. The area used for tornado assessment is included on map 5_2 in the Annexes of this report [L. 203] and also in [L. 204] and [L. 205].

2.4.2.2.2 Snow storm

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 1.6

The snow storm refers to a strong wind, which whirls and carries snow, often accompanied by snowing, storms or hailstones. It is a dangerous phenomenon that occurs mainly in North America [L. 206]. This phenomenon is not observed in the Czech Hydrometeorological Institute network; it rarely occurs in this geographical latitude and is not significant for the purposes of this report [L. 203].

2.4.2.2.3 Dust and sand storm

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 1.6. During the dust storm (sand storm) a strong wind carries fine-grain sand, dust soil, clay or peat into the far distance from the centre of the wind erosion. These storms most frequently occur in arid or semiarid regions. This phenomenon is not observed in the Czech Hydrometeorological Institute network [L. 203].

2.4.2.2.4 Tropical storm

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], note 9 to article 5.14

Tropical storm refers to the third stage of tropical cyclones (the fourth stage is a hurricane). This phenomenon does not occur in Central Europe, since the tropical cyclone in mild latitudes gradually transforms during its life cycle into an extratropical cyclone, which is a common local phenomenon included in the described meteorological phenomena.

2.4.2.2.5 Drought

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 1.6

For the purposes of this report "drought" is defined as an uninterrupted interval of days with daily precipitation not exceeding 2 mm inclusive (i.e. a period with strong deficit of moisture). Based on the assessment of results of different methods for estimation of extreme drought durations, while considering the data series length, the Czech Hydrometeorological Institute determined the final estimation for the time of repetition of 100 and 10,000 years, see Tab. 64 (for background documentation see report [L. 203], table 2.3.1).

Tab. 64 Recommended estimations of uninterrupted interval with daily precipitation up to 2 mm inclusive with repetition time of 100 and 10,000 years

Parameter	Estimation method	Repetition time	
		100 years	10,000 years
The length of uninterrupted interval with daily precipitation up to 2 mm inclusive (number of days)	MLEEV0	66	105

2.4.2.2.6 Hoarfrost

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 1.6

The following table contains average, maximum and minimum number of days with hoarfrost for the whole observation period at the Temelín and Kocelovice stations. The number of days with hoarfrost also takes into consideration days with hoarfrost (grained) and with transparent hoarfrost.

Tab. 65 Average, maximum and minimum monthly and annual number of days with hoarfrost at the Temelín and Kocelovice stations.

Station	Observation period	Statistics	Month												Year
			1	2	3	4	5	6	7	8	9	10	11	12	
Temelín	1989-2010	Average	6.8	3.4	1.5	0.2	0.0	0.0	0.0	0.0	0.0	0.9	2.7	6.0	21.4
		Maximum	20.0	12.0	4.0	2.0	1.0	0.0	0.0	0.0	0.0	4.0	8.0	12.0	33.0
		Minimum	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.0	5.0
Kocelovice	1975-2010	Average	6.3	3.0	0.9	0.2	0.0	0.0	0.0	0.0	0.0	0.9	3.4	6.7	21.4
		Maximum	15.0	13.0	6.0	2.0	0.0	0.0	0.0	0.0	0.0	3.0	11.0	15.0	33.0
		Minimum	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.0	6.0

2.4.2.2.7 Hailstones

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 1.6

The following table Tab. 67 taken from [L. 203] contains average, maximum and minimum number of days with hailstones for the whole observation period at the Temelín and Kocelovice stations.

Tab. 66 Average, maximum and minimum monthly and annual number of days with hailstones at the Temelín and Kocelovice stations.

Station	Observation period	Statistics	Month												Year
			1	2	3	4	5	6	7	8	9	10	11	12	
Temelín	1989-2010	Average	0.0	0.0	0.1	0.2	0.4	0.5	0.2	0.0	0.0	0.0	0.0	0.0	1.5
		Maximum	0.0	0.0	2.0	2.0	2.0	2.0	2.0	1.0	1.0	0.0	0.0	0.0	4.0
		Minimum	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Kocelovice	1975-2010	Average	0.0	0.1	0.1	0.2	0.7	0.4	0.4	0.1	0.1	0.0	0.0	0.0	2.1
		Maximum	1.0	1.0	1.0	2.0	3.0	2.0	2.0	1.0	1.0	0.0	1.0	1.0	5.0
		Minimum	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.0

2.4.2.2.8 Lightning

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], article 5.37 – 5.40.

The following table contains average, maximum and minimum number of days with storm for the period 1989-2010 at the Temelín station. According to the article results [L. 207] the annual average number of days with storm approximately correspond to the average annual number of days with occurrence of at least 2 flashes of lightning into the ground in the area of 10 and 15 km.

Values from the Temelín observatory are considered as the most representative for the given purpose regardless of the fact that the time series covers only 21 years, since the data are local data and based on a comparison with surrounding stations there is no sufficiently tight link of storm characteristics of individual stations, as described in more detail in the following paragraph. Furthermore, the observation of storm activities also includes phenomena that are significantly remote from the station, as described in Section 2.5. 4 reports [L. 203].

For the sake of comparison, the table Tab. 67 contains the number of days with storm at the Tábor, České Budějovice, Kocelovice and Temelín stations. The figure Fig. 13 represents the fluctuation of the number of days with storm at these stations.

Tab. 67 Average, maximum and minimum monthly and annual number of days with storm.

Station	Observation period	Statistics	Month												Year
			1	2	3	4	5	6	7	8	9	10	11	12	
Temelín	1989 - 2010	Average	0.1	0.1	0.7	1.3	4.5	6.4	6.5	4.6	1.0	0.0	0.0	0.0	25.2
		Maximum	1.0	1.0	4.0	5.0	9.0	12.0	11.0	11.0	4.0	1.0	0.0	0.0	39.0
		Minimum	0.0	0.0	0.0	0.0	1.0	1.0	1.0	1.0	0.0	0.0	0.0	0.0	17.0
Kocelovice	1975 - 2010	Average	0.1	0.1	0.5	1.9	6.3	7.6	7.8	7.1	1.6	0.1	0.1	0	33.2
		Maximum	1	2	3	5	15	12	13	13	6	1	1	0	45
		Minimum	0	0	0	0	1	2	2	3	0	0	0	0	19
Tábor	1961 - 2010	Average	0.1	0.1	0.4	1.1	4.1	5.4	5.4	4.4	1.3	0.1	0	0.1	22.5
		Maximum	2	1	2	4	7	12	11	10	5	1	0	1	35
		Minimum	0	0	0	0	0	1	0	1	0	0	0	0	12
České Budějovice	1961 - 2010	Average	0.1	0.2	0.4	1.5	4.4	6.6	6.7	6	1.6	0.3	0.1	0.2	28.1
		Maximum	1	2	2	5	10	13	14	14	7	4	1	3	51
		Minimum	0	0	0	0	1	1	1	1	0	0	0	0	13

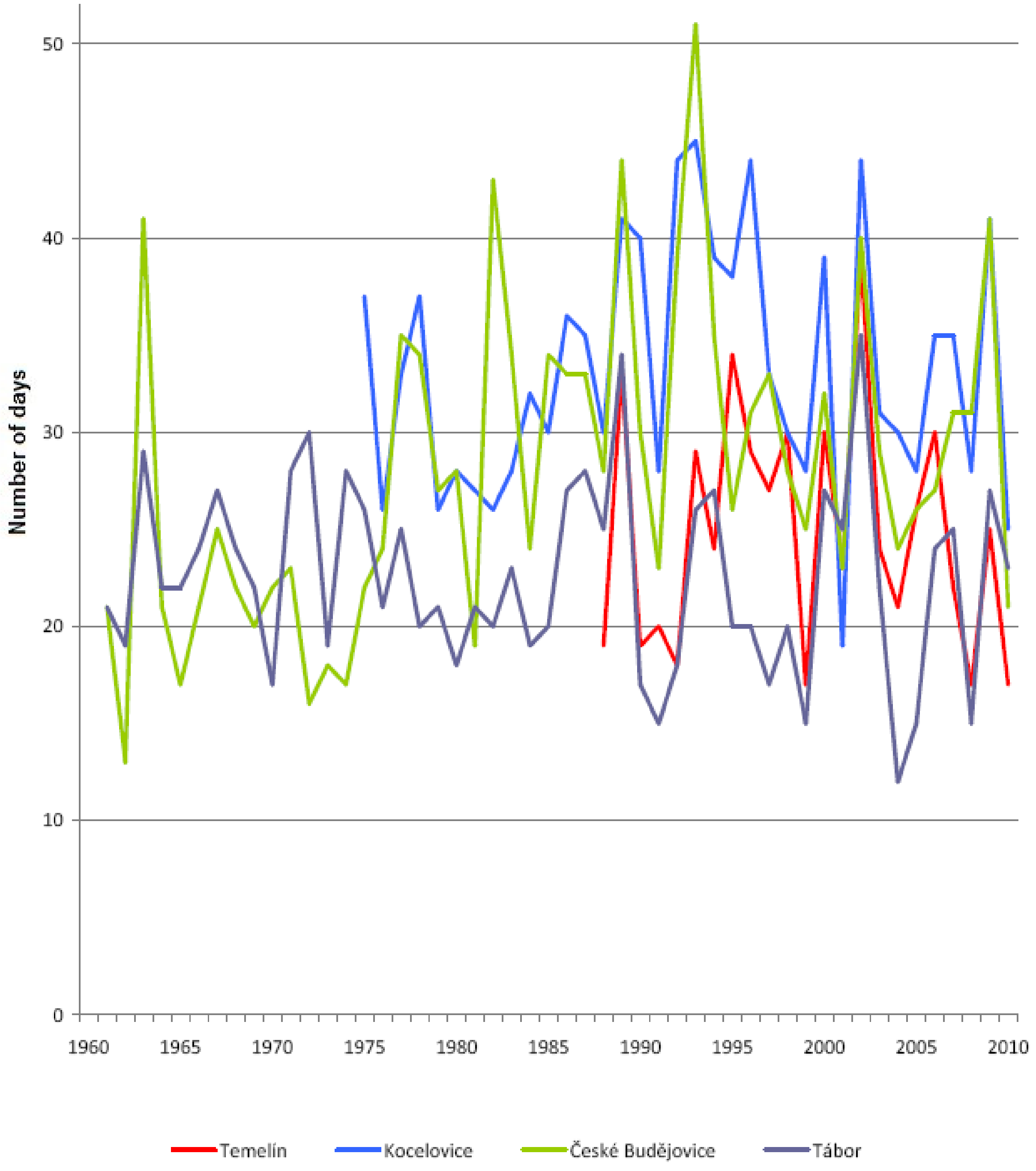

Fig. 13 Fluctuation of the annual number of days with storm at the Temelín, Kocelovice, Tábor and České Budějovice stations.

As follows from the figure and table, the average number of days with storm is highest at the Kocelovice station and lowest at the Tábor station. These differences might be caused by the location of the station and different length of the observation period; the different type of station also plays an important role. An important factor for observing phenomena is the observer quality; in long-term average the number of observed phenomena at volunteer stations are therefore usually lower than at professional stations.

In accordance with Section 2.3.3 of the report [L. 47] a cloud-earth discharge can be expected during lightning occurrence in Central Europe, between 10 and 15 discharges per square kilometre per year, i.e. one discharge in 10 years for a 100 m radius. At the same time only 1% of these cloud-earth discharges reaches the discharge current of 200 kA. Interferences caused by a lightning discharge can be categorized based on whether it is a direct lighting strike into the lighting protection system (LPS) of a building or indirect lightning strike near air or underground distribution lines. Calculations connected with this phenomenon can be performed only after concrete installation conditions are submitted, i.e. building height, LPS design or characteristic data on air or underground distribution lines.

The impact of lightning occurrence in the Temelín locality (both with regards to the frequency and discharge current) on building structures and system of the power plant can be solved within the EMC conception in the contractor documentation and does not place any limits on the use of the locality for the power plant location.

2.4.2.2.9 Tornados

The phenomenon was processed in accordance with IAEA NS-G-3.4 [L. 12], articles 2.1 and 5.2 - 5.11.

Table 2.7.3 in the Annexes of the Czech Hydrometeorological Institute report [L. 203] contains all tornados that occurred in the Czech Republic and that are documented in historical sources. The table includes information on the occurrence date, phenomenon duration, place and trajectory of occurrence, caused damage and phenomenon intensity on the Fujita scale[21]. In the Czech Republic there were 71 documented tornados in the period of one thousand years, out of which 27 tornadoes had the intensity of F0, 24 ranked into class F1, 18 into F2 and 2 had the intensity of F3. It may be stated that in the last thousand years there has been no documented tornado in the range of approximately 25 km from Temelín NPP; in the 40 km range from Temelín NPP there has been no tornado with an intensity higher than F0.

Within the last ten years the average annual number of documented tornados in the Czech Republic ranges between 3-5 occurrences. In previous years the frequency of documented occurrences was significantly lower. The population, however, was significantly different from today's population; the observance network of meteorological stations providing reliable data has been available only for approximately 100 years; also the methods and intensity of observing and documentation have changed dramatically. Therefore it is not yet possible to conclude that the current intensity of tornado occurrence is higher only due to the increased intensity of observance or if it is also caused by climatic changes.

The calculated probability of tornado occurrence with the intensity of F1 in the Temelín NPP locality is approximately six times in 10,000 years and the probability of tornado occurrence with the intensity of F2 is approximately five times in 1,000,000 years (the first line in table Tab. 68) and the probability of occurrence of the most intense tornado in this intensity class is once in 1,000,000 years (the second line in table Tab. 68). The probability of occurrence of a tornado with intensity higher than F2 in the Temelín locality is less than once in a million years.

[21] The tornado intensity in this report is measured on the Fujita scale. Characteristics of this scale are included in table 2.7.2 of the Annexes of the Czech Hydrometeorological Institute report [L. 203].

Initiating parameters of tornados ranked in class F2 in accordance with IAEA No. 50-SG-S11A [L. 17] are documented in table Tab. 68 taken from the report [L. 203]. Relations from [L. 17] obviously cannot be applied for less intense tornados (class F1 and lower limit of class F2).

Tab. 68 Initiating tornado parameters

Intensity	Translation velocity estimate	Maximum wind velocity (m/s)	Translation velocity (m/s)	Maximum rotary wind velocity (m/s)	Maximum rotary velocity radius /m)	Air pressure drop (hPa)	Air pressure drop rate (hPa/s)
F2	average	36	7	29	50	40	2
	Maximum	67	13.1	54			10

Note 1.: The average estimate of translation velocity is calculated as average translation velocity of all tornados in the given class listed in the Annexes of this report [L. 203]; maximum estimate of translation velocity is calculated as the maximum velocity of tornados in the given class.

Note 2: Total air pressure drop in tornados is not known for the Czech Republic and therefore it was adapted from the most similar tornado types from table 1 recommendations US NRC [L. 215] for region III.

Flying object generated by a tornado

Tornados with the intensity of F2 mostly belong to the lower half of the class in the Czech Republic and are manifested by uprooted trees, extensive windbreaks in forest, roofs torn away from buildings, collapse of gables from brick buildings, total destruction of less stable buildings, turned-over cars and unfinished constructions are usually damaged beyond repair.

The tornado carries loose objects lying on the ground and objects "ripped up" by its destructive force; similarly the whirlwind sucks in objects, which then rotate in the tornado and, if in the right position at the edge of the whirlwind, can be thrown up to a distance of several tens of metres. Available sources for the Czech Republic provide only a little information about the effects of such generated "projectiles" that can cause serious damages. One such case is documented in this locality - on 27 June 1997 a 5 kg hammer was thrown to a distance of approximately 30 m. Other cases document flying objects (e.g. roof covering, parts of branches, etc.).

Size of the tornado trace

For the documented cases the size varies significantly within a wide range. The trace width range starts from several tens of metres for weaker tornados and can reach up to hundreds of metres for tornados with the intensity of F2 and F3.

The trace length is also very variable and it may be stated that as opposed to the trace width there is no explicit relation between the trace length and tornado intensity. It ranges from tens, mostly hundreds of metres to several km, rarely even tens of km. Only in the case of 11 May 1910 could the tornado trace reach approximately 150 km; however, it is not clear if it was one or several tornados.

Tornados usually travel from SW or NW to NE or SE. There are practically no tornados travelling from the east. However, there is still quite a significant variability and the tornado path is usually not documented for older cases.

2.4.2.3 METEOROLOGICAL PARAMETERS INFLUENCING DISPERSION

2.4.2.3.1 Definition of terms

The definitions are processed based on requirements stipulated in Sections 2.23 to 2.27 of IAEA NS-G-3.2 [L. 10].

Meteorological parameters influencing dispersion and background documentation for the locality diffusion model consist of:

- the atmosphere stability category. The stability category also determines the turbulence conditions and vertical temperature layers
- data on the flow vector
- data on precipitation

Dispersion conditions in the atmosphere listed in IAEA NS-G-3.2 [L. 10] may be collectively expressed by stability classes.

2.4.2.3.2 Stability category

Relative frequency of stability classes is included in table Tab. 69.

Tab. 69 Relative frequency of stability classes on the Temelín station

Average frequency in the period	Category						Total
	A	B	C	D	E	F(G)	
	strongly unstable (labile)	unstable	slightly unstable	indifferent (neutral)	stable	strongly stable	
1989-2008	1.1	5.5	15.1	41.8	15.2	21.4	100.0
2000-2010	15.2	8.2	12.3	29.9	10.5	23.9	100.0

Values listed in table Tab. 69 were processed for the period between 1989-2008 and 2000-2010 using different methods, see [L. 203].

2.4.2.3.3 Wind direction and velocity

Wind direction and velocity were processed in accordance with requirements stipulated in IAEA NS-G-3.2 [L. 10], articles 2.18 to 2.22.

Mean wind velocity for individual stability classes depending on the flow direction is listed in table Tab. 70 (average for the period between 2000–2010, i.e. currently used methodologies for atmosphere stability assessment).

Tab. 70 Mean wind velocity for individual stability classes depending on the flow direction. Average for the period 2000-2010

Sector	Category A.F	Category A	Category B	Category C	Category D	Category E	Category F
N	3.357	4.284	2.670	2.719	3.168	2.562	1.711
NNE	3.005	3.861	2.871	3.021	3.263	2.658	1.942
NE	2.636	2.85	2.514	2.787	3.085	2.644	2.024
ENE	2.160	2.210	2.220	2.347	2.613	2.352	1.567
E	2.799	2.325	2.643	3.288	3.700	2.612	1.654

Sector	Category A.F	Category A	Category B	Category C	Category D	Category E	Category F
ESE	4.071	3.432	3.825	4.391	4.543	2.86	1.816
SE	3.364	3.088	2.868	3.666	3.922	2.465	1.611
SSE	2.404	2.374	2.369	2.815	2.895	2.265	1.409
E	1.972	2.097	2.071	2.489	2.277	2.004	1.325
SSW	2.055	2.144	2.035	2.397	2.562	2.157	1.492
SW	2.532	2.669	2.254	2.605	3.353	2.529	1.765
WSW	3.575	3.377	3.792	4.175	4.392	2.717	1.788
W	4.539	2.685	3.445	4.925	5.704	3.021	1.870
WNW	4.571	3.924	4.395	5.111	4.963	3.130	1.854
NW	3.417	3.956	3.273	3.886	3.857	2.697	1.748
NNW	3.370	4.349	3.055	3.236	3.343	2.751	1.782
Total	3.294	3.353	3.208	3.809	4.332	2.680	1.771

Tab. 71 specifies class interval and class wind velocity.

Tab. 71 Specification of class velocity

Class	I	II	III	IV	V	VI	VII	VIII	IX	X	XI	XII
Velocity [m/s]	0-0.1	0.2-1.0	1.1-2.0	2.1-3.0	3.1-4.0	4.1-6.0	6.1-8.0	8.1-12	12.1-16	16.1-20	20.1-25	>25.1
Class velocity	0.05	0.6	1.5	2.5	3.5	5	7	10	15	18	22.5	25.1

Tab. 72 contains relative frequency of wind in individual wind velocity classes listed in table Tab. 71, without distinguishing stable categories for individual months and the year total.

Tab. 72 Relative wind direction frequencies for each wind velocity class at the Temelín station without distinguishing stability classes Period between 2000-2010.

Cat. A - F	**Wind velocity class**												
Sector	I	II	III	IV	V	VI	VII	VIII	IX	X	XI	XII	Total
N	0.070	0.289	0.638	0.863	0.781	0.947	0.267	0.054	0.003	0.000	0.000	0.000	3.912
NNE	0.077	0.379	1.095	1.595	1.152	0.946	0.244	0.050	0.001	0.000	0.000	0.000	5.540
NE	0.122	0.592	2.136	2.93	1.646	0.857	0.131	0.026	0.001	0.000	0.000	0.000	8.443
ENE	0.222	0.82	1.704	1.438	0.714	0.366	0.064	0.010	0.000	0.000	0.000	0.000	5.339
E	0.090	0.681	1.339	1.276	0.863	0.803	0.257	0.077	0.005	0.000	0.000	0.000	5.392
ESE	0.068	0.386	0.853	1.174	1.147	1.682	0.788	0.417	0.032	0.001	0.000	0.000	6.547
SE	0.076	0.354	0.748	0.969	0.752	0.843	0.325	0.122	0.013	0.001	0.000	0.000	4.203
SSE	0.077	0.371	0.732	0.646	0.350	0.262	0.061	0.018	0.003	0.000	0.000	0.000	2.520
S	0.088	0.489	0.878	0.52	0.218	0.140	0.029	0.007	0.000	0.000	0.000	0.000	2.369
SSW	0.132	0.82	1.97	1.143	0.448	0.281	0.061	0.016	0.000	0.000	0.000	0.000	4.872
SW	0.193	1.193	4.094	3.159	1.283	1.125	0.36	0.126	0.006	0.000	0.000	0.000	11.539
WSW	0.159	0.968	2.705	2.313	1.472	1.971	1.081	0.684	0.097	0.011	0.001	0.000	11.461
W	0.143	0.615	1.359	1.70	1.82	3.391	2.001	0.941	0.096	0.013	0.002	0.000	12.083
WNW	0.085	0.378	0.732	1.020	1.231	2.515	1.346	0.567	0.066	0.006	0.001	0.000	7.947
NW	0.073	0.313	0.689	0.941	0.766	0.858	0.336	0.118	0.010	0.002	0.000	0.000	4.105

Cat. A - F	Wind velocity class											
NNW	0.060	0.261	0.608	0.864	0.775	0.837	0.24	0.079	0.004	0.000	0.000	0.000	3.729
Total	1.735	8.907	22.281	22.553	15.42	17.824	7.591	3.313	0.338	0.034	0.005	0.000	100.00

Relative wind direction frequencies for each wind velocity class at the Temelín station for individual stability categories contain detailed tables that are available in the Annexes of the Czech Hydrometeorological Institute report [L. 203].

2.4.2.3.4 Turbulence indicators in the atmosphere

The phenomenon was processed in accordance with the requirements stipulated in articles 2.23 - 2.27 of IAEA NS-G-3.2 [L. 10]. Stability classes, see Section 2.4.2.3.2, contain information about the turbulence rate in the atmosphere, which is required for the calculation of the locality diffusion models. Therefore no additional turbulence indicator is required.

2.4.2.3.5 Precipitation and humidity

The phenomenon was processed in accordance with the requirements stipulated in article 2.28 of IAEA NS-G-3.2 [L. 10].. Results are shown in table Tab. 73.

Tab. 73 Frequency distribution of precipitation intensity in distinguished classes

Class		I	II	II	IV	V	VI	VII	VIII	IX	X	Total
Precipitation intensity [mm/h]		0	0.0-0.1	0.2-0.4	0.5-1.0	1.1-2.0	2.1-3.0	3.1-6.0	6.1-10	10.1-20.0	>20	>=0
Year	2000	91.51	2.68	1.22	1.03	0.71	0.40	0.81	0.54	0.57	0.53	100
	2001	88.98	2.34	1.20	1.11	1.19	0.75	1.36	1.13	1.18	0.76	100
	2002	91.05	1.78	0.88	0.81	0.66	0.52	0.96	0.75	1.16	1.43	100
	2003	94.87	2.65	0.84	0.66	0.41	0.19	0.26	0.06	0.03	0.03	100
	2004	94.95	2.05	1.08	0.84	0.58	0.21	0.21	0.05	0.02	0.01	100
	2005	94.35	3.93	1.31	0.30	0.05	0.03	0.01	0.01	0.01	0.00	100
	2006	92.44	1.88	2.33	1.78	0.93	0.27	0.22	0.08	0.04	0.03	100
	2007	92.98	2.01	2.23	1.43	0.72	0.25	0.28	0.04	0.05	0.01	100
	2008	93.28	2.20	2.22	1.23	0.62	0.19	0.18	0.03	0.02	0.03	100
	2009	87.78	3.51	5.89	1.56	0.79	0.19	0.16	0.06	0.03	0.03	100
	2010	91.20	2.78	2.71	1.80	0.72	0.26	0.32	0.07	0.05	0.09	100
Average		92.13	2.53	1.99	1.14	0.67	0.30	0.43	0.26	0.29	0.27	100

2.4.2.3.6 Air temperature

The phenomenon was processed in accordance with requirements stipulated in IAEA NS-G-3.2 [L. 10], articles 2.33 to 2.35.

Data on air temperature (in the sense of IAEA NS-G-3.2 [L. 10] articles 2.33 and 2.35) are recommended in this section with regards to the influence on dispersion in the atmosphere, focusing mainly on data on vertical layers. Relevant temperature data are implicitly included in the stable class category according to Pasquill, therefore these data do not need to be provided in the dispersion studies.

2.4.2.3.7 Inversion

The phenomenon was processed in accordance with requirements stipulated in IAEA NS-G-3.2 [L. 10], articles 2.23 to 2.27.

Analysis of the inversion frequency was processed based on meteorological measurements from 48 stations of the Czech Hydrometeorological Institute located in the Czech Republic (Tab. 74) between 1 January 1961 and 31 March 2012, regardless of the observation completeness. The meteorological parameters were measured by professional stations with hourly measurements. The following aspects were used for the calculation: the number of lowest observed layers of clouds and the base height and wind velocity. Furthermore, data on the height of the Sun above the horizon were also used.

Tab. 74 contains (from the left) name of the station, geographical location, start and end of observation and number of processed cases (hours with observation).

Tab. 74 List and location of stations used for the calculation of the average inversion frequency in the Czech Republic

Station	Altitude (m)	Longitude (°)	G. Latitude (°)	Beginning	End	Number of cases
Bechyně	409	14.4708	49.3	01/01/1961	10/07/1992	140319
Brno	241	16.6889	49.1531	01/01/1961	31/03/2012	314760
Červená	749	17.5419	49.7772	01/01/1961	31/03/2012	294151
České Budějovice	420	14.4292	48.9469	01/01/1961	30/06/1994	162651
České Budějovice	394	14.4714	48.9519	01/01/1961	31/03/2012	27891
České Budějovice	395	14.4714	48.9519	04/01/2000	31/01/2008	76834
Doksany	158	14.1703	50.4583	01/01/1995	31/03/2012	150347
Dukovany	400	16.1344	49.0958	01/10/1995	31/03/2012	142338
Holešov	222	17.5697	49.3206	01/01/1961	31/03/2012	203696
Cheb	483	12.3911	50.0686	01/01/1961	31/03/2012	324026
Chotusice	235	15.3858	49.9419	01/01/1961	31/03/2012	194399
Churáňov	1118	13.615	49.0683	01/01/1961	31/03/2012	325350
Karlovy Vary	603	12.9143	50.2016	17/04/1961	31/03/2012	299035
Kocelovice	515	13.8408	49.4669	01/03/1975	31/03/2012	277447
Kopisty	240	13.6233	50.5442	27/04/2000	31/03/2012	26331
Kostelní Myslová	569	15.4392	49.16	01/01/1961	31/03/2012	312438
Košetice	534	15.0797	49.5733	01/01/1989	31/03/2012	111626
Kuchařovice	334	16.0864	48.8825	01/01/1961	31/03/2012	301855
Liberec	398	15.0242	50.77	01/01/1961	31/03/2012	314242
Luká	510	16.9533	49.6522	01/10/1974	31/03/2012	163112
Lysá Hora	1322	18.4478	49.5461	01/01/1961	31/03/2012	317600
Mariánské Lázně	540	12.7183	49.9272	01/07/1995	01/01/1999	21936
Milešovka	833	13.9314	50.5547	01/01/1961	31/03/2012	208386
Mošnov	250	18.1217	49.6983	01/01/1961	31/03/2012	314776
Pardubice	225	15.7406	50.0161	01/01/1961	31/03/2012	192357
Pec pod Sněžkou	824	15.7288	50.6921	01/09/1988	31/03/2012	144972
Pilsen	360	13.2694	49.6764	01/01/1961	31/12/1981	60085
Pilsen	360	13.3786	49.765	01/07/2004	31/03/2012	67860
Pilsen	343	13.3833	49.7431	04/01/2000	30/06/2004	38280
Polom	748	16.3228	50.3519	01/01/2006	31/03/2012	54720
Pouchov	243	15.8433	50.2456	01/01/1961	31/12/1981	59639
Praděd	1490	17.23	50.0828	01/01/1961	15/09/1997	75463

Station	Altitude (m)	Longitude (°)	G. Latitude (°)	Beginning	End	Number of cases
Praha	364	14.2578	50.1008	01/01/1982	31/03/2012	253368
Praha	282	14.5425	50.1217	01/01/1961	31/03/2012	196107
Praha	232	14.4277	50.0693	01/09/2002	31/03/2012	83670
Praha	302	14.4469	50.0081	01/01/1982	31/03/2012	247733
Přerov	203	17.4064	49.4239	01/01/1961	31/03/2012	192755
Přibyslav	530	15.7625	49.5828	01/01/1961	31/03/2012	208153
Přimda	742	12.6781	49.6694	01/01/1961	31/03/2012	279289
Sedlec	474	16.1206	49.1708	01/01/1961	31/03/2012	194680
Sněžka	1602	15.7403	50.7358	18/10/2008	01/01/2012	22525
Svratouch	737	16.0336	49.735	01/01/1961	31/03/2012	313205
Šerák	1328	17.1086	50.1875	01/01/2004	31/03/2012	71709
Temelín	503	14.3419	49.1978	01/01/1989	31/03/2012	200964
Tušimice	322	13.3281	50.3767	01/01/1982	31/03/2012	265009
Ústí nad Labem	375	14.0411	50.6839	01/01/1979	31/03/2012	270850
Ústí nad Orlicí	402	16.4222	49.9803	01/01/1961	31/03/2012	160594
Zatec	273	13.5775	50.3789	01/01/1961	31/12/1981	53648

2.4.2.4 MEASURING STATIONS AND THEIR INSTRUMENTATION

Input data for assessed meteorological parameters were obtained from the measuring network of the Czech Hydrometeorological Institute. The station selection for individual indicators was based on articles 2.12 to 2.31 of IAEA NS-G-3.2 [L. 10] and on the experience of the Czech Hydrometeorological Institute from its operation.

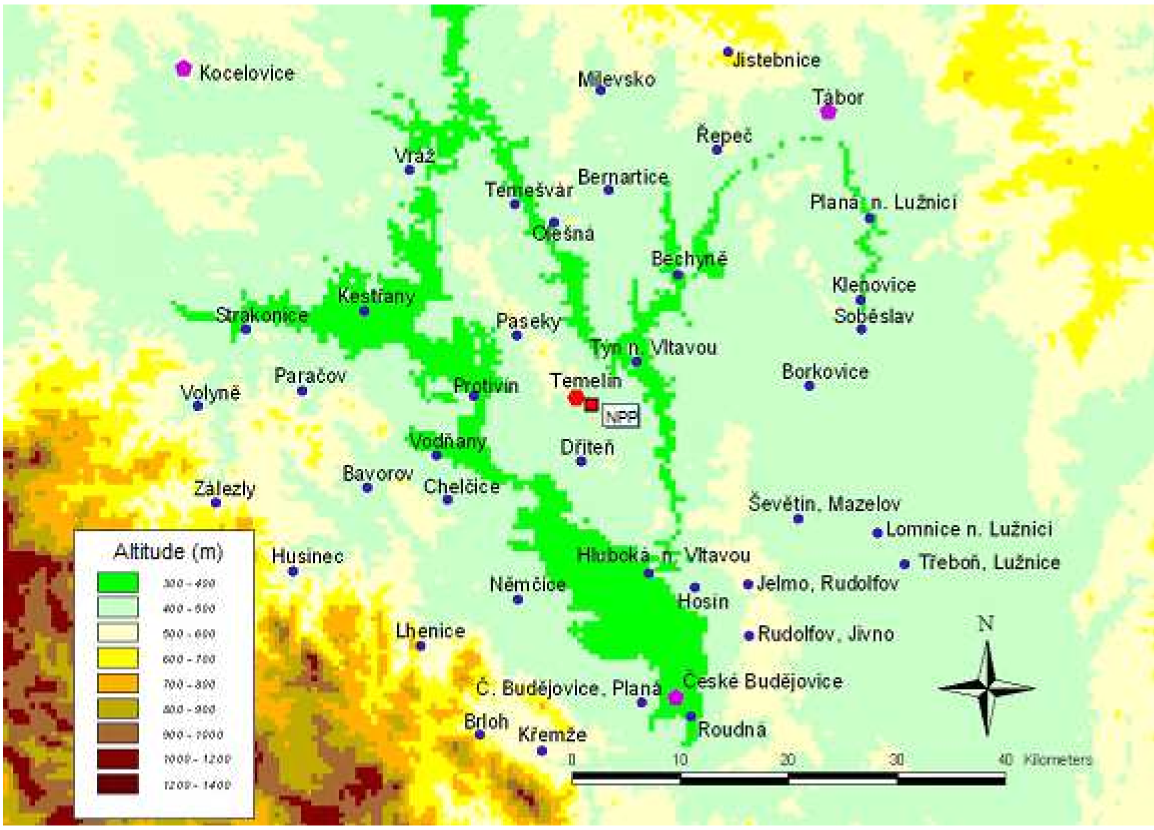

Fig. 14 Location of stations used for the analysis of climatic situations within the distance of 40 km from Temelín NPP

Tab. 75 Stations used for the analysis of climatic situations within the distance of 40 km from Temelín NPP

Station	Distance from Temelín (km)	Altitude (m above sea level)	Used for the characterization											
			Temperature	Wind	Precipitation	Snow	Snow storm	Dust and sand storm	Drought	Hoarfrost	Hailstones	Lightning	Tornados	Dispersion
Bavorov	21	436				X	X	X					X	
Bechyně	15	409				X	X	X					X	
Bernartice	19	467				X	X	X					X	
Borkovice	22	419				X	X	X					X	
Brloh	32	582				X	X	X					X	
České Budějovice Budějovice, Planá	29	420	X			X	X	X					X	
České Budějovice Budějovice, pob.	27	394				X	X	X	X			X	X	
Dříteň	6	420				X	X	X					X	
Hluboká n. Vlt.	17	376				X	X	X					X	
Hosín	20	488				X	X	X					X	
Husinec	30	492				X	X	X					X	
Chelčice	15	466				X	X	X					X	
Churáňov	55	1004		X		X	X	X					X	
Jelmo, Rudolfov	23	505				X	X	X					X	
Jistebnice	35	580				X	X	X					X	
Kestřany	21	372				X	X	X					X	
Klenovice	27	421				X	X	X					X	
Kocelovice	47	519	X	X	X	X	X	X	X	X	X	X	X	
Křemže	32	524				X	X	X					X	
Lhenice	27	558				X	X	X					X	
Lomnice n. Luž.	30	423				X	X	X					X	
Milevsko	28	442				X	X	X					X	
Němčice	19	435				X	X	X					X	
Olešná	16	452				X	X	X					X	
Paračov	25	498				X	X	X					X	
Paseky	8	483				X	X	X					X	
Planá nad Lužnicí	32	406				X	X	X					X	
Praha, Ruzyně	101	364		X		X	X	X					X	
Protivín	9	394				X	X	X					X	
Roudné	31	393				X	X	X					X	
Rudolfov, Jivno	26	558				X	X	X					X	
Řepeč	26	475				X	X	X					X	
Soběslav	27	421				X	X	X					X	
Strakonice	31	426				X	X	X					X	
Šěvětín, Mazelov	23	438				X	X	X					X	

Station	Distance from Temelín (km)	Altitude (m above sea level)	Used for the characterization											
			Temperature	Wind	Precipitation	Snow	Snow storm	Dust and sand storm	Drought	Hoarfrost	Hailstones	Lightning	Tornados	Dispersion
Tábor	35	459	X		X	X	X	X	X			X	X	
Temelín	0	503				X	X	X					X	
Temešvár	19	421				X	X	X					X	
Třeboň, Lužnice	33	420			X	X	X	X					X	
Týn n. Vltavou	7	371				X	X	X	X				X	
Vodňany	14	395				X	X	X					X	
Volyně Nihošovice	35	448				X	X	X					X	
Vráž	26	436				X	X	X					X	
Zálezly	34	569				X	X	X					X	

In accordance with IAEA NS-G-3.4 [L. 12], article 3.1, the instrumentation and its installation of measuring stations that obtained the data is designed exclusively in accordance with WMO requirements and following methodological regulations of the Czech Hydrometeorological Institute (e.g. [L. 214]). The technological equipment of individual stations and observation times and methods are therefore comparable at all stations. For extreme wind information on wind-measuring equipment and their location on individual stations is included in the Annexes of the Czech Hydrometeorological Institute report [L. 203]. The Annex contains photographic documentation of the wind velocity measurement and used meteorological stations - fig. 7.1 to 7.4 and table 5_3.

2.4.3 REQUIREMENTS AND CRITERIA

Tab. 76 Requirements and criteria for the assessment of meteorological aspects of the locality.

ID	Criteria requirement in Decree No. 215/1997 Coll.		Requirements in IAEA NS-R-3	
5.1	Article 5i)	Extremely unfavourable conditions for the dispersion of atmospheric discharges, mainly due to the morphology of the close vicinities.		
5.2			3.8	The extreme values of meteorological variables and rare meteorological phenomena listed below shall be investigated for the site of any installation. The meteorological and climatological characteristics for the region around the site shall be investigated.
5.3			3.9	In order to evaluate their possible extreme values, the following meteorological phenomena shall be documented for an appropriate period of time: wind, precipitation, snow, temperature and storm surges.

ID	Criteria requirement in Decree No. 215/1997 Coll.		Requirements in IAEA NS-R-3	
			3.10	The output of the site evaluation shall be described in a way that is suitable for design purposes for the plant, such as the probability of exceedence values relevant to design parameters. Uncertainties in the data shall be taken into account in this evaluation.
5.4			3.11	*Lightning* The potential for the occurrence and the frequency and severity of lightning shall be evaluated for the site.
5.5			3.12	*Tornadoes* The potential for the occurrence of tornadoes in the region of interest shall be assessed on the basis of detailed historical and instrumentally recorded data for the region.
			3.13	The hazards associated with tornadoes shall be derived and expressed in terms of parameters such as rotational wind speed, translational wind speed, radius of maximum rotational wind speed, pressure differentials and rate of change of pressure.
			3.14	In the assessment of the hazard, missiles that could be associated with tornadoes shall be considered.
5.6			3.15	*Tropical cyclones* The potential for tropical cyclones in the region of the site shall be evaluated. If this evaluation shows that there is evidence of tropical cyclones or a potential for tropical cyclones, related data shall be collected.
			3.16	On the basis of the available data and the appropriate physical models, the hazards associated with tropical cyclones shall be determined in relation to the site. Hazards for tropical cyclones include factors such as extreme wind speed, pressure and precipitation.
			3.17	In the assessment of the hazards, missiles that could be associated with tropical cyclones shall be considered.
5.7			4.1	A meteorological description of the region shall be developed, including descriptions of the basic meteorological parameters, regional orography and phenomena such as wind speed and direction, air temperature, precipitation, humidity, atmospheric stability parameters, and prolonged inversions.

ID	Criteria requirement in Decree No. 215/1997 Coll.		Requirements in IAEA NS-R-3	
			4.2	A programme for meteorological measurements shall be prepared and carried out at or near the site with the use of instrumentation capable of measuring and recording the main meteorological parameters at appropriate elevations and locations. Data from at least one full year shall be collected, together with any other relevant data that may be available from other sources.

2.4.4 DOCUMENTS PROVIDING A BASIS FOR THE ASSESSMENT

- Czech Hydrometeorological Institute – Tornáda na území ČR a Slovenska (Tornados on the Territory of the Czech Republic and the Slovak Republic) http://old.chmi.cz/torn/Tornada na uzemi CR a Slovenska [L. 204]
- Czech Hydrometeorological Institute – Savé víry (Suction Vortexes) http:/old.chmi.cz/torn/poznamky/saveviry [L. 205]
- Meteorologický slovník výkladový a terminologický (Meteorological terminological glossary), page 45, Academia and Ministry of the Environment of the Czech Republic. 594 s. ISBN 80-85368-45-5, Praha, 1993 [L. 206]
- NOVÁK, P., ŽEJDLÍK, T., TOLASZ, R.: Deset let využívání dat detekce blesků v Českém hydrometeorologickém ústavu (Ten Years of Use of Data Obtained from Detection of Lightning in the Czech Hydrometeorological Institute). Meteorological Reports, volume 62, p. 165-172, 2009 [L. 207]
- IAEA No. 50-SG-S11A Extreme Meteorological Events in Nuclear Power Plant Siting, Excluding Tropical Cyclones, Safety Guides, Vienna, 1981[L. 17]
- IAEA NS-G-3.2 Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Safety Guide, Vienna, 2002 [L. 10]
- AEA NS-G-3.4 Meteorological Events in Site Evaluation for Nuclear Power Plants, Safety Guide, Vienna, 2004 [L. 12]
- US NRC Regulatory Guide 1.76 Design–basis Tornado and Tornado Missiles for Nuclear Power Plants, 2007 [L. 215]
- REISS R-D., THOMAS M. (1997): Statistical Analysis of Extreme Values. Page 95-107. Birkhäuser Verlag, 1.ed., Basel; Boston; Berlin [L. 208]
- TRUPL J.: Intensity krátkodobých dešťů v povodích Labe, Odry a Moravy (Intensities of Short-Term Rains in the Elbe, Oder and Morava River Basins). Works and studies, issue 97. Water Research Institute Prague, 1958 [L. 209]
- Projekt měřícího systému pro meteorologickou stanici Dukovany (Measuring System Project for Meteorological Station Dukovany), ZM Hasoft spol. s r.o., 1992 [L. 210]

- Zpráva k ročnímu vyhodnocení meteorologických pozorování na observatořích ČHMÚ Dukovany a Temelín za rok 2000, 2001, 2002, 2003, 2004, 2005 (Report on Annual Evaluation of Meteorological Observations at Observatories of the Czech Hydrometeorological Institute Dukovany and Temelín for 2000, 2001, 2002, 2003, 2004, and 2005), Czech Hydrometeorological Institute [L. 211]
- Vybrané meteorologické údaje pro JE Temelín (Selected Meteorological Data for Temelín NPP), Czech Hydrometeorological Institute, 2009 [L. 212]
- Vybrané meteorologické údaje pro JE Temelín (Selected Meteorological Data for Temelín NPP), Czech Hydrometeorological Institute, 2010 [L. 213]
- ŽÍDEK D., LIPINA P.: Návod pro pozorovatele meteorologických stanic (Instructions for Observers at Meteorological Stations), Guideline No. 13, Czech Hydrometeorological Institute, Ostrava, 2003 [L. 214]
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- Adam Vizina, Petr Vyskoč: Zadávací bezpečnostní zpráva (ZBZ): Interpretace kritérií pro umísťování jaderných zařízení (Initial Safety Analysis Report (ISAR): Interpretation of Siting Criteria of Nuclear Installations), T. G. Masaryk Water Research Institute, 04/2012 [L. 253]
- SÚJB BN-JB-1.14, Interpretace kritérií pro umísťování jaderných zařízení a návrh jejich průkazů (Interpretation of criteria for the siting of nuclear installations and a proposal for their evidence documentation), SÚJB, April 2012 [L. 268

2.4.5 METHODS APPLIED TO THE EVALUATION

2.4.5.1 USED TIME SERIES

The used data were obtained by measuring and observing at stations in the monitoring network of the Czech Hydrometeorological Institute that are archived in the Clidata digital database. A repeated data revision was also performed within the processing of the data. The selection of series was done in accordance with requirements of IAEA NS-G-3.4 [L. 12], article 3.7, which prefers at least three years of measuring. The Temelín observatory has currently only 22-year data series (or shorter for some elements). Therefore sufficiently long time series were used from surrounding stations, depending on the assessed meteorological phenomenon, measurement and monitoring quality of the given station. With the exception of the water rate of snow, estimates of extreme values were always calculated from the series of one station and estimates calculated from several stations were then assessed by an expert to determine the final recommended value For water rate of snow several series were compiled, which were constructed as the global seasonal maximum for the given range from Temelín. This procedure appeared to be the most suitable with regards to the variability of this type of data. It should be noted that IAEA NS-G-3.2 [L. 10] and IAEA NS-G-3.4 [L. 12] do stipulate recommended procedures for series length, however a different approach is allowed depending on the specific situation.

2.4.5.2 METHODS FOR ESTIMATING EXTREME VALUES

Estimates were calculated using software equipment described in the study [L. 208]. The calculation used a model with classical Gumbel distribution (two-parameter distribution, designated as MLEEV0), as well as three modifications of estimates, using the so-called unified model for extreme values (GEV). These three methods are referred to in this report as LRSEV, MLEEV and SSEEV. These models are described in more detail in [L. 208].

Generally speaking, out of the four stated statistical models (Gumbel, LRSEV, MLEEV, SSEEV) the most plausible results are usually yielded by the Gumbel distribution model SSEEV0. SSEEV yields low estimates and finally models MLEEV and LRSEV show signs of certain instability in estimates of hundred-year values. Used Gumbel distribution of probability (Type I) depends on two parameters: parameter representing location (μ - mu) and parameter representing dispersion (σ - sigma).

Distribution function of standard Gumbel distribution is given by the following relationship:

$$G_0(x) = \exp(-e^{-x}). \tag{2.4}$$

By adding location and dispersion parameters μ and σ you get the Gumbel model (EV 0):

EV 0: $\{G_{0,\mu,\sigma} : \mu \text{ real}, \sigma>0\}$, where

$$G_{0,\mu,\sigma}(x) = \exp(-e^{-(x-\mu)/\sigma}) \text{ and } G_{i,\alpha,\mu,\sigma}(x) = G_{i,\alpha}\left(\frac{x-\mu}{\sigma}\right), \text{ i = 1 and 2.} \tag{2.5}$$

Estimates of parameters μ and σ using the MLEEV0 method were calculated numerically by smallest square method using the iteration procedure. The iteration equation is available in [L. 208]. Gumbel distribution is also recommended by IAEA No. 50-SG-S11A [L. 17] and IAEA NS-G-3.4 [L. 12].

Values in tables listing estimated extreme values were processed for a confidence interval, with mean estimate of extreme values for the given repetition time at the significance level 5.

2.4.5.3 METHODS FOR ESTIMATING PARAMETERS OF RARE PHENOMENA

Methods for estimating rare meteorological phenomena draw on Section 5.1 of IAEA NS-G-3.4 [L. 12].

Tab. 77 Methods for estimating rare meteorological phenomena [L. 203]

Meteorological phenomena	Assessment method
Snow storm	Frequency analysis
Dust and sand storm	Frequency analysis
Drought	Assessment of drought was carried out by calculating the number of continuous days with daily precipitation up to 2 mm including the assessment of extremely long annual intervals of drought using methods for estimating extreme values (see Section 2.4.5.2 of this report).
Hoarfrost	Frequency analysis (the number of days with hoarfrost also takes into consideration days with hoarfrost (grained) and with transparent hoarfrost - see Section 2.4.2.2.6 of this report).
Hailstones	Frequency analysis (hailstones are round, cone-shaped or irregular pieces of ice with diameters larger than 5 mm. Description of hailstones and of this phenomenon observation is included in [L. 214]).
Lightning	Frequency analysis of the number of days with storms (this number also includes days with near storms, distant and very distant storms [L. 214]).
Tornados	Frequency analysis of tornado occurrence and parameters, calculation of initiating tornado parameters in accordance with IAEA No. 50-SG-S11A [L. 17] based on information on the occurrence date, phenomenon duration, place and trajectory of occurrence, caused damage and phenomenon intensity, as included in the Czech Hydrometeorological Institute database. Assessed initiating parameters were compared with initiating parameters of tornados included in [L. 215], which also used the source of data on pressure drop that are otherwise unavailable from Czech observing.

2.4.5.4 METHODS FOR ESTIMATING METEOROLOGICAL CONDITIONS INFLUENCING DISPERSION

2.4.5.4.1 Stability condition factors

The phenomenon was processed in accordance with requirements stipulated in IAEA NS-G-3.2 [L. 10], Sections 2.23 to 2.27. Frequency distribution into the following 3 factors was processed separately:

- atmosphere stability classes according to Uhlig-Pasquill (implicitly also contains turbulence and temperature layers)
- wind vectors (direction and speed) with and without regard to the stability class
- precipitation

2.4.5.4.2 Stability classes

Classification of stability classes is currently carried out automatically in the Temelín observatory based on results of the AMS measurement according to software for calculating stability classes using the Pasquill scale with sampling in 10-minute intervals in accordance with regulations for operating nuclear power plants.

Other parameters were used to determine the stability class, such as vertical gradient of air temperature or radiation data measured on samples in 10-minute intervals at the Temelín observatory (place of measurement C7TEME01).

Between 1989-2008 stability classes were assessed using the Uhlig-Pasquill method [L. 210] based on synoptic observation. Summarized results are included in the report [L. 203]. Since 2008 some synoptic observations are not realized during the night and therefore the stated methodology for stability class classification is not suitable for assessing observations carried out after the specified date.

Data for the period starting from 2000 are based on the above described automatic AMS measurements.

Precipitation measurements in the winter period prior to 2000 did not provide valid data for the given indicator, therefore data from 2000 - 2010 were used. Background data for the period between 2000 and 2005 are hourly precipitation values measured by AMS Vaisala on the C1TEME01 station; for the period 2006 - 2010 these data consisted of precipitation intensity produced by automatic measuring station Vaisala, weather status module, with sampling in 10-minute intervals (place of measurement C7TEME01).

2.4.5.4.3 Wind vector

Frequency analysis of background data in order to calculate the annual relative frequency of selected wind velocity classes in relation to the wind direction and atmosphere stability categories and without distinguishing atmosphere stability. Frequency analysis of background data on precipitation within class intervals of precipitation intensities required by the submitter.

For stable wind roses the result of annual average relative frequency calculations for wind direction observance in the Temelín observatory were used for individual atmosphere stability classes, as well as without distinguishing stability classes, with or without distinguishing wind speed classes. Average values for the given period were calculated from the annual values.

2.4.5.4.4 Precipitations

Annual frequency distribution of precipitation from 2000 to 2005 was taken from [L. 210]. Data for this period were calculated in the Temelín observatory based on hourly precipitation data. Annual frequency distribution of precipitation between 2006-2010 was calculated in the Temelín observatory from precipitation intensities measured in 10-minute intervals by AMS Vaisala. Average values for the period between 2000-2010 were then calculated from the annual values.

2.4.5.4.5 Inversion

With regards to the fact that objective measurements cannot be used to calculate the average temperature gradient in the lower atmosphere layer for the Czech Republic, the assessment used Stability Class calculation according to Pasquill in the Turner modification. The used calculation method for stability classes corresponds to Sections 6.3 and 6.4 of the U.S. guide. Environmental Protection Agency [L. 248]. The number of cases of bad dispersion conditions was calculated both as a summary for the whole Czech Republic, and for the Temelín locality (Temelín station), as the sum of class occurrence frequencies. The relative frequency of E and F classes (slight and medium stability) with regards to the total number of cases was calculated on the basis of these results. The ratio of relative frequency for Temelín and of the average for the Czech Republic was then used as the final criterion.

2.4.6 DEFINITION OF THE AREA EXAMINED

2.4.6.1 METEOROLOGICAL STATIONS USED FOR ASSESSMENT OF THE TEMELÍN LOCALITY

Areas for the analysis of meteorological parameters were determined in accordance with requirements stipulated by IAEA NS-G-3.4 [L. 12], Section 2.2. Priority was given to measurements and observation data from the nearest stations, alternatively from the nearest stations with sufficiently long time of high-quality observation and located at a similar altitude and in similar orographic terrain. Stations used for the analysis of individual phenomena are listed in the description of corresponding sections on individual phenomena.

Overview of geographic background data (location, altitude) of all used stations is included tables 5_1 to 5_3 in the Annexes of this report [L. 203]. Their location is clear from map No. 5_1 and 5_3 of the Annexes of the above-cited report. A list of stations used for processing individual characteristics by chapters is included in tab. Tab. 75. The area used for tornado occurrence is specified on map 5_2 in the Annexes of the report and contains all stations in this area and all places besides stations in this area. Station selection was processed in accordance with recommendations stipulated in Section 3.10 of IAEA NS-G-3.4 [L. 12].

The Temelín observatory is situated in close proximity to the Temelín power plant and it is equipped by an extensive measuring programme and above-standard technical equipment; its aim is to monitor the local climate of the Temelín power plant. The aim also determined the location of the observatory, which is representative of the local climate near the power plant. In order to assess extreme values of meteorological phenomena the longest available data series from the relevant areas for the required characteristics were used and their expert evaluation in relation to the Temelín locality was processed. For more details see [L. 203]. Average climatic characteristics values stated in this report are included in the Czech Hydrometeorological Institute report [L. 212] and [L. 213], including correlations between basic and local data.

The examined area for inversion assessment is the locality within 3 km from the plot border of Temelín NPP (the Temelín NPP observatory is the place of measurements), which is then compared with average values for the whole Czech Republic.

2.4.6.2 EXTREME VALUES

Tab. 78 Definition of the area examined for extreme values of meteorological phenomena

	Assessed area
Temperature	The Temelín observatory (C1TEME01) was used for the assessment (data from climatic terms, data from extreme thermometers and hourly data), as well as the three nearest stations with the required length and quality of observation. These stations include the secular station in České Budějovice and Tábor (data from climatic terms and data from extreme thermometers) and Kocelovice (data from climatic terms, data from extreme thermometers and hourly data). Location of stations is listed in tab. 5_2 and on map 5_3 in the Annexes of this report [L. 203].
Wind velocity	The selection was conditioned by the measuring software and meteorological station equipment, measurement quality and length of time series, distance from Temelín and the geographic location. For processing extreme wind gusts the following professional stations, located within 100 km from Temelín NPP with terrain comparable to the Temelín locality, were used: Praha-Ruzyně, Kocelovice, Churáňov and Temelín (see map in Fig. 15). Additional information on the above mentioned meteorological stations is included in table 5_3 in the Annexes of this report [L. 203]).
Precipitation	Available stations with data series on precipitation in minute or hour duration with a length of at least 18 years, i.e. Temelín, Kocelovice, Tábor and Třeboň. The location of stations is listed in tab. 5_1 and on map 5_3 in the Annexes of this report [L. 203].
Snow conditions	Precipitation measuring and climatic stations of the Czech Hydrometeorological Institute within the range of 35 km from Temelín NPP.

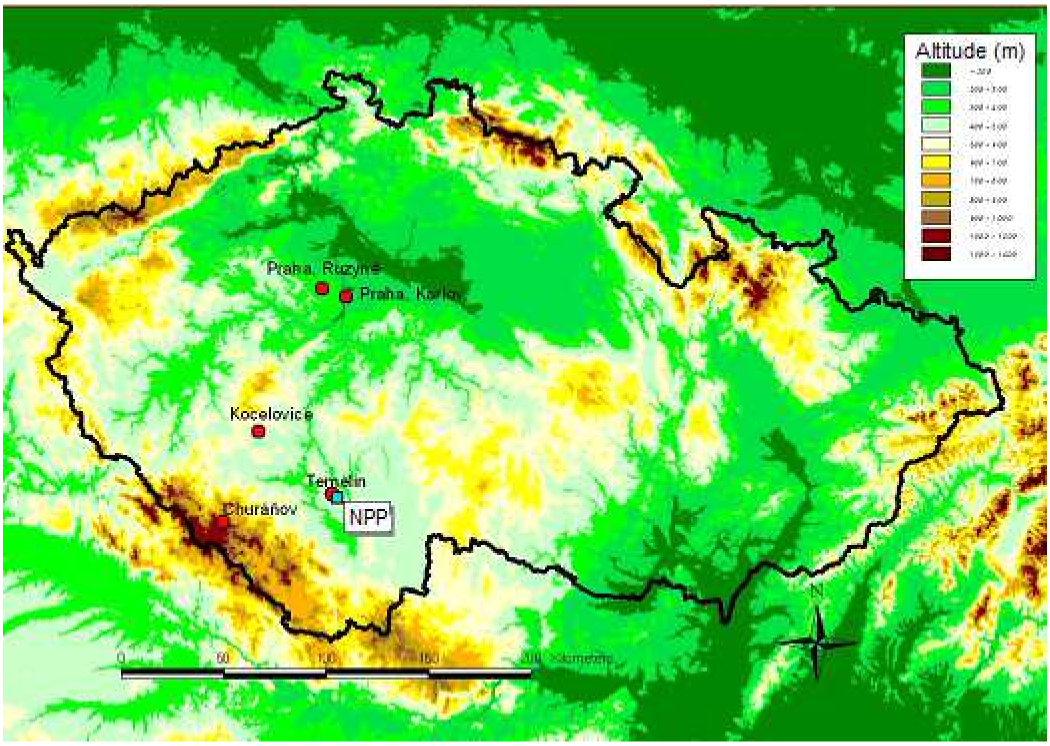


Fig. 15 Location of stations used for the wind analysis.

2.4.6.3 RARE METEOROLOGICAL PHENOMENA

An assessment area was selected for each rare meteorological phenomena in accordance with specific phenomenon observation requirements (see [L. 203]). A list of assessment areas is included in table Tab. 79.

Tab. 79 Delimitation of the assessment area for rare meteorological phenomena

	Assessed area
Snow storm	Czech Republic
Dust and sand storm	Czech Republic
Drought	Precipitation measuring stations in the closest vicinity of Temelín with long observation series were selected for the assessment. These stations are Tábor (35 km to the northeast), České Budějovice (26 km to the south southeast), Kocelovice (47 km to the northwest), Týn n. Vltavou (7 km to the east northeast from Temelín NPP). The location of stations is described in detail in tab. 5_1 in the Annexes and on map 5_3 in the Annexes of this report [L. 203].
Hoarfrost	Data observed at professional stations in Temelín and Kocelovice were used for the assessment of this phenomenon. Location of stations is listed in detail in tab. 5_1 and on map 5_3 in the Annexes of this report
Hailstones	Data observed at professional stations in Temelín and Kocelovice were used for the assessment of this phenomenon. The location of stations is listed in detail in tab. 5_1 and on map 5_3 in the Annexes of this report
Lightning	Data observed at professional stations in Temelín and Kocelovice were used for the assessment of this phenomenon. The

	Assessed area
	observation of storm activities on the stations also includes phenomena that are significantly remote from the station, as described in Section 2.6.4. The locations of stations are listed in detail in tab. 5_2 and on map 5_3 in the Annexes of this report [L. 203].
Tornados	The IAEA recommendations do not include any concrete recommendations on the selection of an area for tornado analysis. The US NCR recommendation [L. 215] suggests considering areas with the size of approximately 100,000 km^2 to determine tornado parameters. Due to the location of Temelín NPP, such area would include Bavaria and Austria to the southeast and south; there are no available data on tornado occurrence for these areas. For the above mentioned reasons and with regards to the path of documented tornados, the area used for the calculation covers the Pilsen region and South Bohemian region, the counties of Pelhřimov, Havlíčkův Brod, Kutná Hora and also part of the Central Bohemian region located south of the 50th parallel (approximately 25,000 km2). Used tornado cases are listed in table 2.7_4, which form a part of the Annexes of this report [L. 203]. The analysed area is depicted on a map in Fig. 16 of this report.

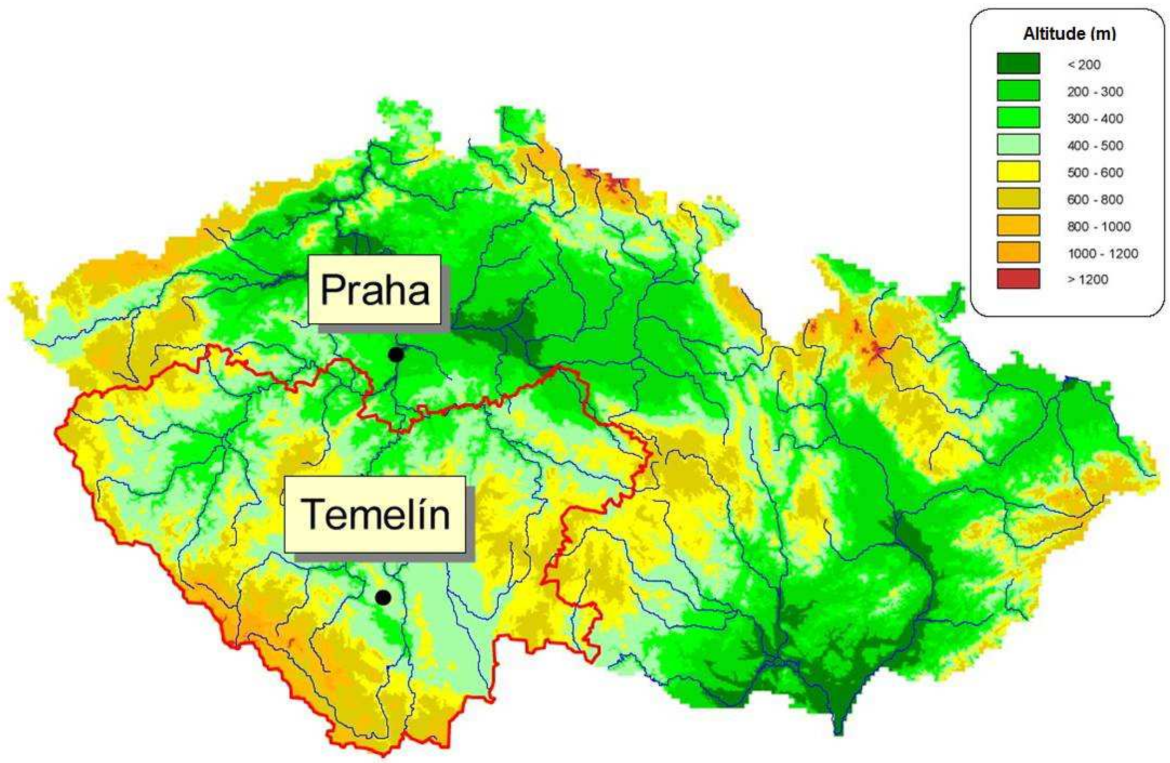


Fig. 16 The area of interest for tornado occurrence. Borders are marked in red.

2.4.6.4 METEOROLOGICAL PARAMETERS INFLUENCING DISPERSION

The listed group of background documentation was created based on measurements in the Temelín observatory of the Czech Hydrometeorological Institute.

2.4.7 DETAILED ASSESSMENT OF ALL REQUIREMENTS AND CRITERIA SPECIFIED BY DECREE NO. 215/1997 COLL. IN COMBINATION WITH THE IAEA NS-R-3 STANDARD

2.4.7.1 CRITERION DEFINED BY ARTICLE 5 PAR. I) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Article 5 paragraph i) of Decree No. 215/1997 Coll. [L. 1] is reproduced in Tab. 76 under item 5.1. The meaning of the criterion, as interpreted in [L. 268], is to exclude localities with extremely unfavourable atmospheric conditions for dispersion of possible radioactivity leaks. The criterion does not apply to water vapours from normal operation of the power plant.

The occurrence of extremely unfavourable dispersion conditions follows from regional climatic conditions and from the terrain formation. Specifically it must be documented that the narrower location is not situated in a mountain or sub-mountain valley with slope height exceeding 200 m, nor in a basin with extremely high occurrence of inversions. The extreme occurrence refers here to values three times higher than the average for the Czech Republic.

Based on the morphology of the narrower Temelín location as described in Section 2.1.2.1 the Temelín site is a plain on which the plant building site is located at an elevation of 3 to 10 m above the surrounding ground. This location is predisposed to good air motion.

The following Tab. 80 lists the relative frequency of individual stability categories according to Pasquill both in the Temelín locality, and on average for the Czech Republic, including the average calculated for the period between 1 January 1961 and 31 March 2012, regardless of the observation length or measuring interruptions on individual stations. The last column includes the frequency of stable layers (it is the sum of E and F categories)

Tab. 80 Relative frequency (%) of stability categories according to Pasquill.

Area	Strongly unstable	Moderately unstable	Slightly unstable	Neutral	Slightly stable	Moderately stable	Stable	Total
	A	B	C	D	E	F	E+F	A to F
Temelín	0.40	6.44	11.10	33.44	24.44	24.17	48.61	100.00
Total for the Czech Republic	0.42	6.32	11.33	41.47	16.40	24.07	40.47	100.00

As follows from Tab. 80, the ratio of category frequencies with stable layers (sum of E and F categories) in the Temelín locality, i.e. a locality with unfavourable dispersion conditions, is 1.20 times the average for the Czech Republic. This value is lower than the value of criterion No. 3, i.e. the Temelín locality meets the dispersion condition criterion.

From the viewpoint of Article 5 par. i) of Decree No. 215/1997 [L. 1], the Temelín locality is not limited by increased occurrence of extremely unfavourable dispersion conditions, as follows from:

- terrain morphology, which does not have the characteristics of a valley or basin with unfavourable conditions for the dispersion of atmospheric discharges
- frequency of unfavourable dispersion meteorological conditions, which is lower than three times the average of the Czech Republic

2.4.7.2 REQUIREMENTS IN SECTION 3.8 OF IAEA NS-R-3

The text of the requirement in Section 3.8 of IAEA NS-R-3 [L. 6] is reproduced in table Tab. 76 under item 5.2.

Extreme meteorological phenomena listed in Section 3.8 of IAEA requirements [L. 6] were analysed based on data series measured on the Temelín observatory of the Czech Hydrometeorological Institute and other stations with comparable meteorological conditions in the surroundings. The list of stations used for obtaining data for individual indicators is included in table Tab. 75.

Rare meteorological phenomena were analysed in accordance with requirements stipulated in Sections 3.8 to 3.17 of IAEA NS-R-3 [L. 6] and the results are provided in Sections 2.4.2.1 and 2.4.2.2 of this report. The list of meteorological stations used for obtaining data for individual indicators is also included in the above-cited table Tab. 75.

2.4.7.3 REQUIREMENTS IN SECTIONS 3.9 AND 3.10 OF IAEA NS-R-3

The text of the requirement in Sections 3.9 and 3.10 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 76 under item 5.3.

The extreme meteorological phenomenon values are presented in accordance with requirements stipulated in Sections 3.9 and 3.10 of IAEA NS-R-3 [L. 6] as extreme values, which are not expected to be exceeded in the reference period (repetition time) used for concrete meteorological phenomenon (see Tab. 81).

Tab. 81 Reference period (repetition time) for determining extreme values of meteorological phenomena

Meteorological phenomenon	Reference period [year]	Number of table with the values
Temperature	10,000	Tab. 59
Wind velocity	10,000	Tab. 60
Precipitation	100	Tab. 61
Snow conditions	10,000	Tab. 63

Notes to the extreme values of meteorological parameters are included in Sections 2.4.2.1.1 to 2.4.2.1.4 of this report. These values will form part of design basis of the ETE3,4 project (see Section 2.10.3 of this report).

2.4.7.4 REQUIREMENTS IN SECTION 3.11 OF IAEA NS-R-3

The text of the requirement in Section 3.11 of IAEA NS-R-3 [L. 6] is reproduced in table Tab. 76 under item 5.4.

The number of days with storm was assessed in accordance with recommendations in IAEA NS-G-3.4 [L. 12]. The results are summarized in table Tab. 67. The following table also contains average, maximum and minimum number of days with storm for the period 1989-2010 at the Temelín station. According to the article results [L. 207] the annual average number of days with storm approximately correspond to the average annual number of days with occurrence of at least 2 flashes of lightning into the ground in the area of 10 to 15 km.

In accordance with Section 2.3.3 of the report [L. 47] a cloud-earth discharge can be expected during lightning occurrence in Central Europe, between 10 and 15 discharges per square kilometre per year, i.e. one discharge in 10 years for a 100 m radius. At the same time only 1% of these cloud - earth discharge currents reach or exceed 200 kA.

Calculations for interferences caused by the lightning discharge can be performed only after concrete installation conditions are submitted, i.e. building height, LPS design or characteristic data on air or underground distribution lines (see Section 2.4.2.2.8 of this report).

2.4.7.5 REQUIREMENT IN SECTIONS 3.12 TO 3.14 OF IAEA NS-R-3 STANDARD

The text of the requirement in Sections 3.12 and 3.14 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 76 under item 5.5.

Tornado occurrence was assessed based on detailed historical data for the last 1,000 years and on data measured by the equipment in the given region during the operation of each station. The reference data source was the Czech Hydrometeorological Institute database on tornado occurrence in the Czech Republic. For the assessment of tornados an area covering 25,000 km^2 of the Czech Republic was used (see map in Fig. 16). Tornado occurrence frequency and their initiating parameters were assessed in accordance with recommendations in IAEA NS-G-3.4 [L. 12] while considering [L. 215]. The final initiating tornado parameters are summarized in table Tab. 68 and will form a part of the ETE3,4 design basis.

Tornado initiating parameters listed in Tab. 68 are comparable with tornado parameters determined for the region III in regulatory guide IAEA RG 1.76 [L. 215]. 3 types of flying objects generated by a tornado are defined for region III in the above-cited standard:

- steel pipe with 168 mm diameter, 4.58 mm long, weighing 130 kg, with the impacting speed of 24 m/s
- passenger car with weight 1178 kg, impacting speed of 24 m/s, impact height 9.14 m at most
- steel sphere with 2.54 cm diameter and the impacting speed of 6 m/s

Due to the fact that the impact of airplane weighing 7 tons at the impacting speed of 200 m/s is already included in initiating events for the Temelín NPP locality (see Sections 2.2.8 and 2.10.4.3 of this report) and safety-related constructions, systems and components will be designed to withstand its effect, the resistance towards the effect of flying objects generated by a tornado can be assumed with a large reserve.

2.4.7.6 REQUIREMENTS IN SECTIONS 3.15 TO 3.17 OF IAEA NS-R-3 STANDARD

The text of the requirements in Sections 3.15 and 3.17 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 78 under item 5.6.

In accordance with requirements stipulated by 3.15 the potential of a tropical storm was assessed in the locality. Section 2.4.2.2.4 of this report explains why a tropical storm cannot occur in this locality. This settled requirements on the locality stipulated in Sections 3.15 to 3.17 of IAEA NS-R-3 [L. 6].

2.4.7.7 REQUIREMENTS IN SECTIONS 4.1 AND 4.2 OF IAEA NS-R-3

The text of the requirements in Sections 4.1 and 4.2 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 76 under item 5.3.

Reference meteorological description of the locality was processed in 1984 in order to obtain the permission for siting ETE1,2 based on the information obtained from the contemporary station network of the Czech Hydrometeorological Institute. The meteorological description of the locality has been continuously updated; since 2000 results from the Temelín meteorological observatory are also included. The model used to process this report meets the requirements of IAEA NS-R-3 [L. 6] and it is based on data time series including the year 2010 [L. 203]; the model is described in detail in Section 2.4.2 of this report. Implementation of the meteorological measuring programme for the Temelín locality is contractually negotiated between the ETE1,2 operator, ČEZ, a.s., and Czech Hydrometeorological Institute.

2.4.8 CONCLUDING ASSESSMENT

Information from the Czech Hydrometeorological Institute issued for the locality so far has been verified and completed in order to assess the meteorological conditions in the Temelín locality in accordance with the methodology included in Section 2.4.5 (for the background material see [L. 203]).

The Temelín locality meets the criterion according to Article 5 par. i) of the Decree No. 215/1997 Coll. [L. 1] The meteorological situation in the Temelín locality has been further analysed with regards to requirements stipulated in Sections 3.8 to 3.14, 4.1 and 4.2 of IAEA NS-R-3 [L. 6] and no meteorological phenomenon excluding or conditioning the construction of ETE3,4 has been discovered.

In accordance with Section 3.10 of IAEA NS-R-3 [L. 6] design bases related to extreme and rare meteorological phenomena have been determined and are included in Sections 2.10.3.1 to 2.10.3.5 of this report.

2.5 HYDROLOGICAL CONDITIONS

2.5.1 SCOPE OF THIS SECTION

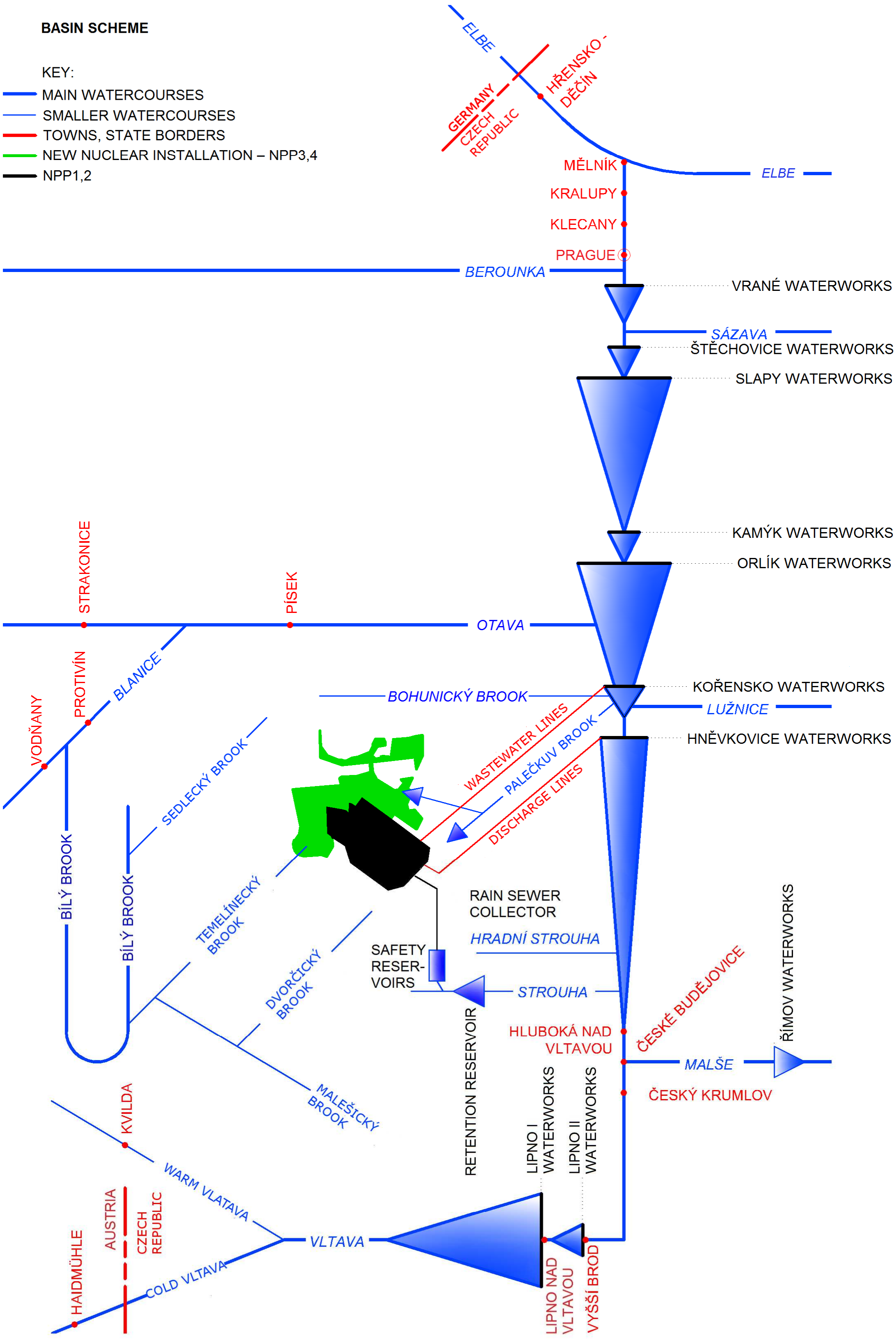
This section describes the circulation of surface water in the locality and its surroundings and assesses the possibility of flooding of the land intended for siting of the nuclear installation as caused by applicable natural forces or by water flood caused by water dam accidents or a combination of both [L. 18]. This section also defines the prerequisites for process water supply of ETE3,4.

2.5.2 SUMMARY OF FACTS

2.5.2.1 HYDROLOGY OF THE AREA

As far as hydrology is concerned, the NPP locality is located on the divide of two rivers, the Vltava and the Bílý brook. The Bílý brook falls within the Blanice basin, into which it flows between the city of Vodňany and Protivín, approximately 13 km away from the area. The Blanice basin has no larger water management significance in relation to the power plant. On the other hands, two waterworks were built on the Vltava for the purposes of Temelín NPP, completing the "Vltava cascade" of water dams. These are the Hněvkovice reservoir and the Kořensko underground facility, located 5 and 6 km from the area. The Hněvkovice reservoir was built on river km 210.39 and is used as a source of raw water. The Kořensko underground facility, located at river km 200.41, is used for homogenization of discharged wastewater. The weir structure of the Kořensko underground facility is flooded by backwater at the maximum level of the Orlík reservoir. Maximum reservoir levels are $H_{MAX,HNĚVKOVICE}$ = 370.10 m above sea level, $H_{MAX,KOŘENSKO}$ = 352.60 m above sea level, $H_{MAX,ORLÍK}$ = 353.60 m above sea level [L. 219]

The scheme of the Vltava basin is represented in Fir. 17.



Fir. 17 Scheme of the Vltava basin

2.5.2.2 THE VLTAVA RIVER

The area of interest falls within the Vltava river basin. The Vltava is one of the main rivers of the Czech river system. The basin covers the overall area of 28,708 km^2. The Vltava river spring is located in the "Šumava" mountains, at the foothills of Černá Hora, at the altitude of 1,172 m above sea level. The Vltava river is a left-bank tributary of the river Elbe, to which it flows at its length of approximately 436 km. The Vltava river is the longest river in the Czech Republic. A system of water damns, the so-called "Vltava cascade" was built in the past on the river [L. 18], [L. 277].

Tab. 82 List of water dams in the "Vltava cascade" [L. 219].

reservoir	[river km]	S_{BASIN} [km^2]	S_{FLOODS} [ha]	$L_{BACKWATER}$ [km]	V_{TOTAL} [mil. m^3]	H_{MAX} [m above sea level]	Q_a [m^3/s]	Q_{100} [m^3/s]
Lipno I.	329.543	948.20	4870.00	42.000	309.502	725.60	13.10	359
Lipno II.	319.108	999.40	45.00	1.500	1.664	562.70	13.40	374
Hněvkovice	210.39	3540.30	276.67	18.650	21.090	370.10	30.60	1054
Kořensko	200.405	7828.90		9.985	2.8	352.60	54.90	1387
Orlík	144.650	12106.00	2732.70	68.000	716.50	353.60	83.50	2180
Kamýk	134.730	12217.90	195.00	19.920	12.98	284.60	83.70	2065
Slapy	91.610	12956.80	1162.60	43.000	269.30	270.60	85.20	2250
Štěchovice	84.318	12992.80	95.70	7.380	10.40	219.40	85.25	2290
Vrané	71.325	17784.60	263.00	13.000	11.1	200.10	110.00	2670

The cascade has mainly hydro energy and water management significance, as well as recreational. The total output of hydropower stations is 761.5 MW, annual average production of peak energy is 1,160 GWh. As far as water management is concerned, the reservoir is used mainly to improve the minimum flow and protect the land against floods. As far as recreation goes, it is used as a waterway of regional importance, which joins the navigable Elbe waterway. The Elbe river, as a waterway of European importance, flows into the North Sea near the Hamburg dock in Germany [L. 18].

Water reservoirs have been built only on Vltava tributaries. The most significant reservoirs include namely Švihov, located on the Želivka river, and the Římov reservoir on the Malše river. The Římov reservoir is used as a source of drinking water for České Budějovice and its surroundings. The Švihov water reservoir, known more commonly as "Želivka", is a source of drinking water for the whole Central Bohemian region, including Prague. It is the largest water reservoir in the Central Europe. Fluctuating flow and changeable water quality of the Vltava river limit the possibility of water withdrawal directly from the watercourse. Only two profiles are used for this purpose. For the Příbram region the withdrawal profile is located in Solenice, at river km 144. For Prague it is a back-up source of drinking water, the "Podolí waterworks", with withdrawal at river km 56.2 [L. 18].

Water reservoir Lipno I is located in Šumava, at the altitude of 725.60 m above sea level. Lipno I is the largest water surface in the Czech Republic, it covers the area of 4,870 ha along state borders with Austria and it is situation in the Protected

Landscape Area and National Park of Šumava. The length of backwater is approximately 42 km and the widest point reaches approximately 5 km.

The dam consists in one third of a concrete gravity dam and in two thirds of a ground rock-filled dam. The average reservoir depth is 6.50 m, the maximum depth at the dam is approximately 25 m. Lower drains are located in the concrete structure: 2x DN 2500 with the capacity of 2x 86.10 m^3/s. Control sluices of safety overflow are located symmetrically above the drains. The dam consists of two 10 m wide flaps with the capacity of 148.42 m^3/s. A busy road leads on the 296 m long crest. For hydro energy purposes, a hydroelectric power station was placed approximately 200 m below the adjacent terrain. There are two Francis turbines in the hydroelectric power station; their installed output is 120 MW, maximum absorption capacity of 2x 46 m^3/s for the fall scope of 161.65 ~ 149.35 m. Turbine inlets consist of two vertical shafts. The drain is made of a waste tunnel with discharge into the equalizing reservoir Lipno II, near the town of "Vyšší Brod". The waste tunnel is about 3.60 km long. The reservoir is used mainly to produce electricity and to regulate water draining situations [L. 219].

Water reservoir Lipno II is located near the town of Vyšší Brod in South Bohemia, in the altitude of 562.70 m above sea level. It is an equalizing reservoir for hydroelectric power plant Lipno I located about 5 km from the reservoir. The dam consists of ground rock-filled dam, adjacent on both sides to the concrete outlet object. The outlet object contains two crest overflows equipped with isolation flaps, one siphon overflow, a gravel gate and the hydroelectric power plant structure. The maximum reservoir depth is 11 m. The total length of the top barrier is approximately 224 m, out of which the concrete outlet object part is 54 m. The continuous hydroelectric power plant is equipped with one Kaplan turbine, which processes the fluctuating flow with the maximum installed capacity of 1.5 MW. The absorption capacity of the turbine is 20 m^3/s at the maximum fall of 9.55 m [L. 219].

The water reservoir Hněvkovice is located about 5 km south from Týn nad Vltavou, at the altitude of 370.10 m above sea level. The total water surface covers the area of 276.67 ha. The flooded area, which is 18.65 km long, extends up to the weir in the town of Hluboká nad Vltavou.

The dam consists of a 23.50 m long concrete gravity dam with three overflows with the width of 3x 12 m. The overflows are enclosed by 7 m high segment closures with the total capacity of 3 x 337 m^3/s, at the surface height of 370.10 m above sea level. The dam crest is used as a public road, which is 191 m long. The altitude of the crest is 372.60 m above sea level. A chamber for boats with a bearing capacity of 300 t is built on the right bank. The ground plan dimension of the chamber is 6.00 x 45.00 m, with a maximum fall of 19.00 m. The dock chamber bypass is also the bottom outlet of the reservoir. The waterworks also contains a small hydroelectric power plant (MVE). MVE contains two Kaplan turbines with the installed output of 9.6 MW and maximum absorption capacity of 2x 30m^3/s, for the fall scope of 9.30 ~ 14.80 m. Continuous or semi-peak operation of MVE takes into consideration the daily equalization of the flow after withdrawals by pumping stations of technological water for Temelín NPP.

The reservoir was built to supply Temelín NPP with technological water. The pumping station of technological water is located on the left bank as a separate object. The pumping station was designed for the withdrawal of the designed 4x 1000 MW, i.e. 1,30~4,16 m^3/s. At the maximum reservoir surface level, i.e. 370.10 m above sea level, the depth of the place of withdrawal is about 17.5 m. During periods with less waters, the required withdrawals are ensured in cooperation with the Lipno water reservoir. The flow is improved if the water level drops below the so-called "dispatching level", whereas the minimum level is 364.60 m above sea level. The Hněvkovice reservoir, provided that sufficient depth is maintained, therefore ensures safe withdrawals under any climatic or operation conditions, be it sediment flow or slush-ice jams [L. 219]. The Hněvkovice waterworks was originally designed for required withdrawal of 4.80 m^3/s for 4x 1,000 MW_e. The current allowed withdrawal for 2x 1,000 MW in accordance with Decision ref. No. OŽP/7497/2006/Si is Q_{mean} 1,800 m^3/s and Q_{max} = 3,000 m^3/s [L. 225].

Overview of basic hydrologic data in the profile of the Hněvkovice water dam, which supplements information stated in table Tab. 82, is listed in tables Tab. 83 to Tab. 85. Data in the following tables are in accordance with ČSN 75 1400 [L. 227].

Tab. 83 Profile characteristics of the "VLTAVA - Hněvkovice waterworks" (Class II)

watercourse:	Vltava	A	P_a	Q_a
hydrological order No.:	1-06-03-076	[km^2]	[mm]	[m^3/s]
in profile:	Hněvkovice waterworks - dam	3540.30	769	30.60

Key:

A [km^2] – basin area

Q_a [m^3/s] – long-term average

P_a [mm] – long-term average precipitation

Tab. 84 M-day flow rates of the "VLTAVA - Hněvkovice waterworks" (Class II) watercourse

M	30	60	90	120	150	180	210	240	270	300	330	355	364
Q_M	71.9	52.0	40.5	33.2	27.6	24.1	21.1	19.5	17.5	15.6	13.6	11.4	10.4

Key:

Q_M [l/s] – M-day flow rates

Tab. 85 N-year flow rates of the "VLTAVA - Hněvkovice waterworks" (Class II) water course

N	1	2	5	10	20	50	100	1000	10,000
Q_N	196	276	409	529	667	874	1,054	1,798	2,600

Key:

Q_N [m^3/s] – N-year flow rates

The **Pořensko** underground facility is located approximately 3.5 km from the town of Týn nad Vltavou, at an altitude of about 352.60 m above sea level. The weir basin was constructed below the confluence of the Lužnice and Vltava rivers, with regards to fluctuating flows caused by the Hněvkovice waterworks operation. The total area of the basin reaches approximately 7,828.90 km^2.

In addition to the water surface stabilization in Týn nad Vltavou, the purpose of this structure is the homogenisation of discharged waste water from Temelín NPP, in order to prevent its stratification in the Orlík water reservoir. The Kořensko waterworks is flooded by increasing the water level of the Orlík reservoir at its maximum level. The total volume of the weir basin is about 2.8 mil. m^3. Average long-term n-year flow rate reaches about 54.9 m^3/s.

The construction length of the reinforced concrete structure of the weir is 89 m. The overflow crest level is 347.80 m above sea level; the level of the upper side of the weir pillars is 353.10 m above sea level. The maximum operation depth of the basin is approximately 9 m. The weir object has four control sluices with hollow flap valves. The weir sluices are 4x 20 m wide with the maximum capacity of 1,110 m^3/s. A dock chamber for boats with a bearing capacity of 300 t is built on the left bank. The ground plan dimension of the chamber is 6.00 x 45.00 m; the dock chamber is without equipment. A small hydroelectric power plant (MVE) is situated between the chamber and the weir object. It is operated in tandem with the Hněvkovice MVE. MVE contains two Kaplan turbines with the installed output of 3.8 MW and maximum absorption capacity of 2x 40m^3/s, for the fall scope of 2.00 ~ 6.20 m. On the left bank under the weir object the so-called "Dumping object" was constructed as a part of Temelín NPP. The hydro-energy potential of waste water discharge from Temelín NPP is used in the dumping object for its own small hydroelectric power plant. Waste waters are subsequently discharged into Kořensko waterworks MVE suctions or through pillars of the weir object evenly by the Vltava river channel profile [L. 219].

Overview of basic hydrologic data in the profile of the Kořensko waterworks is stated in tables Tab. 86 to Tab. 88. The key for these tables is identical with the key provided for table Tab. 83, Tab. 84 and Tab. 85.

Tab. 86 : Profile characteristics of the "VLTAVA - Kořensko waterworks" (Class II)

watercourse:	Vltava	A	P_a	Q_a
hydrological order No.:	1-07-05-001	[km^2]	[mm]	[m^3/s]
in profile:	Kořensko waterworks - dam	7828.90	716	54.90

Tab. 87 M-day flow rates of the "VLTAVA - Kořensko waterworks" (Class II) watercourse

M	30	60	90	120	150	180	210	240	270	300	330	355	364
Q_M	141	93.7	71.5	59	49.4	42.1	37.5	33.6	29.9	25.3	19.6	15.1	12.7

Tab. 88 N-year flow rates of the "VLTAVA - Kořensko waterworks" (Class II) watercourse

N	1	2	5	10	20	50	100	1000	10,000
Q_N	259	366	543	702	882	1,153	1,387	2,347	-

The **Orlík** water reservoir is located in the South Bohemian and Central Bohemian regions at an altitude of 353.60 m above sea level. It is the largest waterworks in the Czech Republic. The flooded area reaches into the area of interest of Temelín NPP.

The Orlík water reservoir is made of a direct concrete gravity dam with the height of 81.50 above the bottom. The dam crest is 361.10 m above sea level and is 450 m long. It is used as a public road. Three crest overflows, 3 x 15 m wide, are used to transfer large volumes of water. The overflows are controlled by segment shutters with the total capacity of 2,183 m^3/s. Two bottom control drains 2x DN 4000, with the capacity of 371 m^3/s, are located symmetrically below the safety overflows. The hydroelectric power plant, with four Kaplan turbines and installed output of 364 MW and maximum absorption capacity of 4x 150 m^3/s, for the fall scope of 45.00 ~ 71.50 m, is situated on the left bank. Fluctuating discharges from the power plant are regulated by the Kamýk equalizing reservoir, built for this purpose lower on the river course. The waterworks also includes equipment for ship transport. Equipment for lifting ships up to 3.5 t and maximum size of 8.50 x 2.60 is situated on the right bank. It consists of a platform truck with a retractor rope. Ships with the bearing capacity up to 300 t are transported using an inclined ship lift, which is currently without equipment. Construction length of the ship lift is 191 m [L. 219].

The **Kamýk, Slapy, Štěchovice and Vrané** waterworks are not in any way linked to Temelín NPP and do not reach into its area of interest.

The above-mentioned waterworks are owned by the state. They are managed by the company Povodí Vltavy, which is a public enterprise, including the Horní Vltava, Dolní Vltava and the Berounka subsidiaries. The subsidiaries manage and maintain significant and designated watercourses and waterworks in the assigned territory. Furthermore, they are responsible for water management equipment, they monitor the quality and quantity of surface and underground waters and make plans of the territory. As part of their function they also create prerequisites and conditions for using the intangible and tangible assets for allowed or authorized purposes. The Povodí Vltavy public enterprise has the right to manage waters in accordance with the conditions stipulated by water authorities.

The area of interest of Temelín NPP falls within the Vltava river basin, the Horní Vltava subsidiary [L. 219].

Activities included within the management of the Vltava basin include channel dredging. This reduces the risk of an abrupt blockage of the channel by sediment deposits.

Tab. 89 Basic hydrological data on water reservoirs in the "Vltava cascade" [L. 219].

RESERVOIR	FLOWS [m^3/s]			RESERVOIR VOLUMES [mil. m^3]				LEVELS [m above sea level]		DAM HEIGHT AND LENGTH [m]		
	Q_a	Q_{335}	Q_{364}	$V_{PROTECTIVE}$	$V_{SERVICE}$	V_{STABLE}	V_{TOTAL}	H_{MAX}	$H_{SERVICE}$	L_{DAMS}	$H_{ABOVE\_TERRAIN}$	$H_{ABOVE\_FOUNDATIONS}$
Lipno I.	13.1	2.65	1.91	33.16	252.99	23.35	309.50	725.6	716.1~724.9	296.00	25.00	42.00
Lipno II.	13.4	3.16	2.06	-	1.44	0.22	1.66	562.7	557.6~562.7	224.00	11.50	19.50
Hněvkovice	30.60	6.43	4.10	-	12.15	8.94	21.09	370.1	364.6~370.1	191.00	23.50	33.50
Orlík	83.50	17.30	11.30	62.07	374.43	280.00	716.50	353.6	329.6~351.2	450.00	81.50	90.50
Kamýk	83.70	17.10	11.40	-	4.65	8.32	12.98	284.6	282.1~284.6	158.00	17.00	24.50
Slapy	85.20	16.10	11.70	-	200.50	68.80	269.30	270.6	246.6~270.6	260.00	60.00	67.50
Štěchovice[22]	85.25	17.00	12.10	-	4.20/3.34	6.24/7.10	10.40	219.4	214.8/215.8~219.40	124.00	22.00	31.00
Vrané	110.00	20.40	14.90	-	2.52	8.58	11.10	200.1	199.1~200.1	96.80	18.00	22.00

[22] The Štěchovice waterworks has different minimum storage levels for shipping and non-shipping seasons. Data included in the table are for non-shipping/shipping season.

Tab. 90 Basic hydrological data of the Vltava river basin [L. 18]

item	flow	hydrometric profile	S_{BASIN} [km^2]	precipitation [mm/year]	drain [mm/year]	ϕ	q_S [$l/s.km^2$]	Q_a [m^3/s]	Q_{100} [m^3/s]
1	Vltava	Vyšší Brod	999.40	922	423	0.46	13.4	13.4	374
2	Vltava	Český Krumlov	1338.31	837	397	0.47	12.59	16.8	440
3	Vltava	nad Malší	1861.74	788	352	0.45	11.14	20.7	545
4	Malše	mouth	979.10	723	230	0.32	7.3	6.92	442
5	Vltava	pod Malší	2840.84	766	307	0.4	9.73	27.6	810
6	Vltava	nad Bezdrev. potokem	3031.40	757	296	0.39	9.39	28.4	854
7	Bezdrevský pot.	mouth	335.62	611	121	0.2	3.82	1.28	122
8	Vltava	pod Bezdrev. potokem	3367.02	742	279	0.38	8.84	29.7	943
9	Vltava	Hluboká n. Vltavou	3450.87	739	276	0.37	8.73	30.1	970
10	Vltava	upstream of the Lužnice	3645.09	733	276	0.36	8.47	30.8	1010
11	Lužnice	mouth	4226.17	667	181	0.27	5.75	24.3	556
12	Vltava	downstream of the Lužnice	7871.26	698	221	0.32	7.01	55.2	1460
13	Vltava	Zvíkov nad Otavou	8196.90	694	216	0.31	6.86	56.2	1530
14	Otava	mouth	3788.22	681	216	0.32	6.86	26.0	1080
15	Vltava	downstream from Otava	11985.49	690	216	0.31	6.86	82.2	2400
16	Vltava	Kamýk	12217.92	687	214	0.31	6.77	82.7	2442
17	Vltava	upstream from Mastník	12541.59	684	210	0.31	6.65	83.4	2468
18	Mastník	mouth	331.44	604	118	0.20	3.73	1.23	105
19	Vltava	downstream from Mastník	12880.74	682	207	0.30	6.57	84.6	2496

item	flow	hydrometric profile	S_{BASIN} [km^2]	precipitation [mm/year]	drain [mm/year]	ϕ	q_S [l/s.km^2]	Q_a [m^3/s]	Q_{100} [m^3/s]
20	Vltava	Štěchovice	13298.33	678	203	0.30	6.43	85.5	2527
21	Sázava	mouth	4349.19	664	183	0.28	5.8	25.2	810
22	Vltava	downstream from Sázava	17658.92	675	198	0.29	6.27	110.7	2970
23	Vltava	Zbraslav	17827.73	673	196	0.29	6.23	110	2970
24	Berounka	mouth	8861.39	586	128	0.22	4.06	36.0	1545
25	Vltava	Modřany	26689.12	644	174	0.27	5.51	147	4114
26	Vltava	upstream from the Únětický brook	27215.19	642	170	0.27	5.51	147	4135
27	Únětický potok	mouth	47.60	506	66	0.13	2.09	0.10	23
28	Vltava	downstream from the Únětický brook	27262.79	642	172	0.27	5.44	148	4138
29	Vltava	mouth	28090.00	638	169	0.29	5.34	149	4172
30	Elbe	upstream from the Vltava	13713.98	697	226	0.32	7.15	98.0	1560
31	Elbe	downstream from the Vltava	41804.95	657	187	0.29	5.94	248	4575

2.5.2.3 REGIONAL WATERCOURSES FALLING WITHIN THE VLTAVA BASIN

Watercourses situated in the Vltava basin form a dense network of rivers. Smaller watercourses usually include rivers with lower amounts of water, which are relatively short and have larger longitudinal slope. Water discharge from the locality is accelerated and highly fluctuating. The Vltava river, on the other hand, possesses relatively large water amounts and marked minimum flow rates. Important tributaries of the Vltava river are the Malše, Lužnice, Otava, Sázava and Berounka [L. 18] and [L. 219]).

Tab. 91 Significant tributaries of the Vltava river

river	S_{BASIN} [km^2]	tributary	river km	to	where
Malše	979	right-bank	240.00	Vltava	in České Budějovice
Lužnice	4,226	right-bank	202.20	Kořensko	near Týn nad Vltavou
Otava	3,788	left-bank	169.10	Orlík	near Zvíkov Castle
Sázava	4,350	right-bank	78.40	Vrané	near Davle
Berounka	8,860	left-bank	63.20	Vltava	near Lahovice

The **Malše** river spring is located at the northeast foothills of the Viehberg mountain near the town of Sandl in Austria. It forms the state border in a 22 km long section and subsequently leaves the border near Dolní Dvořiště. A water reservoir, which is a significant source of drinking water for the České Budějovice region, was built on the river near the village of Římov. Malše is a right-bank tributary to the Vltava river, to which it empties in the centre of České Budějovice. The total length of the water course is 96 km, out of which approximately 89.3 km is located in the Czech Republic. The average annual flow rate is 7.26 m^3/s.

The **Lužnice** river spring is situated on the slopes of Novohradské mountains in Upper Austria, at the altitude of approximately 990 m above sea level. It enters the Czech Republic at river km 146, with the total basin area reaching about 650 km^2. It flows through the Třeboň basin, where it has formed part of the traditional set of ponds since the 16th century. At that time the "Rožmberk" pond, which is the largest pond in the Czech Republic, was built on the Lužnice river. The "Třeboň PLA" was entered into the UNESCO system of biospheric reservations. The Lužnice river turns to the southwest and empties into the Orlík reservoir approximately 33 km from NPP Temelín, near the town of Tábor. More precisely, it is a weir of the Kořensko underground facility, which falls within the area of interest of Temelín NPP. The Lužnice river significantly contributes towards increasing the flow rate in the given profile. It empties into the Vltava river at river km 202.200 as a right-bank tributary. The total watercourse length is 208 km. Average annual flow rate is 24.10 m^3/s.

The **Otava** river is formed by the confluence of two mountain torrents, Vydra and Křemené, near the town of Kašperské Hory in Šumava. The flow has the characteristics of a torrent up to Horažďovice, where the flow slows down. Larger tributaries include Volyňka, Blanice and Lomnice. **Blanice**, as a recipient, is included in the area of interest of Temelín NPP. Otava empties into the Orlík reservoir below Zvíkov castle. The reservoir backwater extends up to the 19.4 river km. The Otava empties into the Vltava river as a left-bank tributary at river km 169.100. The total length of the watercourse is 113 km. The average annual flow rate is 26.00 m^3/s.

The **Blanice** river spring is situated in the Šumava PLA, it flows through Bavorov, Vodňany, Protivín and empties into the Otava river before reaching the town of Písek. The "Husinec" water reservoir was built on the river near the town of Prachatice. Blanice is the recipient of the **Bílý brook** at 21.3 river km; the Bílý brook basin covers the east and southeast side of the Temelín NPP locality. It is the longest tributary of the Otava river with the largest volume of water. The watercourse length is 93.3 river km; the area of basin reaches 860.50 km^2.

The **Sázava** river flows through the Vysočina and Central Bohemian regions. It drains part of Bohemian-Moravian Highlands and the southern part of Central Bohemian Highlands. It carries away the clay soil, which gives it its characteristic colour. The total length of the watercourse is 225 km; the area of the basin is approximately 4349 km^2. The river is a right-bank tributary of the Vrané reservoir, which belongs to the Vltava cascade. Due to its geographic location, the Sázava river cannot affect the flow rate situation of the Vltava river in the area of interest of Temelín NPP.

The **Berounka** river is an important watercourse of the West Bohemia and Central Bohemian regions. The watercourse is formed by the confluence of rivers Radbuza and Mže in the centre of Pilsen. The river flows through the town of Beroun and empties into the Vltava river in the southern part of Prague. Significant tributaries include Úslava, Klabava, Střela and the Rakovnický brook. It empties into the Vltava river as its left-bank tributary at river km 63.4. The total watercourse length is 138.9 km; the area of the basin reaches approximately 8,861 km^2. Due to its geographic location, the Berounka river cannot affect the flow rate situation of the Vltava river in the area of interest of Temelín NPP.

2.5.2.4 REGIONAL WATERCOURSES AND LOCAL WATER SURFACES

2.5.2.4.1 Regional watercourses

The spring area of smaller watercourses, on which pond complexes are built, is located in the proximity of the premises. Thanks to the power plant location at an elevation above the surrounding terrain, the nearby watercourses drain water from its territory. The spring area is located at the average altitude of 490 m above sea level; the power plant elevation above the spring area is 10 m on average. The elevation towards the highest surface level of the Hněvkovice reservoir is 132.5 metres.

Due to high sloping on a relatively short distance, individual watercourses have the character of a torrent. These watercourses flow mainly through unpopulated, mostly forested areas [L. 18] and [L. 219]).

Tab. 92 Significant local torrents

item	watercourse	basin - [river km]	L [km]	S_{BASIN} [km^2]	Q_{MEAN} [l/s]
1	Palečkův Brook	Vltava – 208.151	9	12.14	40
2	Hradní Strouha	Vltava – 212.669	4	4.00	13
3	Strouha	Vltava – 214.118	6.5	13.173	43
4	Temelínecký potok	Bílý potok	5	14.16	48

The north, northeast and east slope below the area belongs to the **Palečkův brook** basin. Small water basins, spreading over 0.5 ha at most, are located on its smaller tributaries on the east side of the area. For instance, the Nový pond or Oběšený pond. The Palečkův brook is a left-bank tributary to the Vltava river, it empties into the Kořensko reservoir at river km 208.151. The altitude difference on its approximately 9 km long course is about 126 m.

Tab. 93 Profile characteristics of the "PALEČKŮV BROOK" (Class IV)

watercourse:	Palečkův Brook	A	P_a	Q_a
hydrological order No.:	1-06-03-077	[km^2]	[mm]	[m^3/s]
in profile:	upstream from the crossing with road II/105 Hluboká n. Vltavou – Týn n. Vltavou	2.73	634	0.012

Tab. 94 M-day flow rates of the "PALEČKŮV BROOK" (Class IV) watercourse

M	30	60	90	120	150	180	210	240	270	300	330	355	364
Q_M	32	20	15	11	9	7	5	4	3	2	1	0.5	0.2

Tab. 95 N-year flow rates of the "PALEČKŮV BROOK" (Class IV) watercourse

N	1	2	5	10	20	50	100	1000
Q_N	1.2	1.7	2.6	3.5	4.5	4.2	7.5	13.9

The **Hradní Strouha** brook springs east from the area and empties into the Hněvkovice reservoir after approximately 4 km, at river km 212.669. The Hradní Strouha flows through the village of Litoradlice.

The southeast slope below the area is drained by the **Strouha** brook, which also includes the Karlovec pond, covering the area of approximately 1.80 ha The Karlovec pond is situated near the area, between Hradní strouha and Dvorčický brooks. The watercourse is used as a recipient of cleaned waste water from the rain sewerage of the Temelín NPP. In order to enable discharge, a retention reservoir was built on the watercourse, which is used to transform the water inflow volume. In the past a cascade of three ponds covering the area of 2.2500 ha was built at the altitude of approximately 450 m above sea level. The cascade consists of the Libivský, Mlýnský and Nový ponds near Býšov. A pond complex was also built on the left-bank tributary of the Strouha brook. The largest pond in the complex is the Hůrecký pond, which covers the area of approximately 4.50 ha. The size of other ponds, namely the Pohrobný pond, Barbora or Starý pond, does not exceed 2.25 ha. The Strouha brook is the left-bank tributary of the Vltava river; it empties into the Hněvkovice reservoir at river km 214.118. On its approximately 6.5 km long course, it gains the altitude of 114 m. The smaller populated areas concerned are Hůrka, Knín and Strouha.

Tab. 96 Profile characteristics of the "STROUHA BROOK" (Class III)

watercourse:	Strouha	A	P_a	Q_a
hydrological order No.:	1-06-03-073	[km^2]	[mm]	[m^3/s]
in profile:	approximately 1.4 km downstream from the Karlovec dam	1.01	637	0.005

Tab. 97 M-day flow rates of the "STROUHA BROOK" (Class III) watercourse

M	30	60	90	120	150	180	210	240	270	300	330	355	364
Q_M	12	7	6	4	3	3	2	2	1	1	0.7	0.3	0.1

Tab. 98 N-year flow rates of the "STROUHA BROOK" (Class III) watercourse

N	1	2	5	10	20	50	100	1000
Q_N	0.30	0.60	1.00	1.80	2.60	3.80	5.00	9.30

The south slope below the area is drained by the **Dvorčický brook** with the Dvorčický pond, covering the area of approximately 1.8 ha. Using a transformer located at the altitude of approximately 457 m above sea level, it feeds the complex of ponds on the Dříteňský brook. The Dvorčický brook is a left-bank tributary of the Malešický brook and with its 5 km length it gains the altitude of approximately 82 m. The Malešický brook is a left-bank tributary of the Temelínecký brook. The above mentioned watercourses fall within the Bílý brook basin. The concerned populated areas are the villages of Kočín and Malešice.

The spring of the **Temelín brook** is situated on the south or southeast border of the area. The power plant site in question includes Location 6, which currently contains a communal and inert waste dump, together with the sediment storage from the cooling water treatment plant. Location 6 is currently equipped with its own water management system, which prevents contaminated water from leaking into the environment. The Temelínecký brook is a left-bank tributary of the Bílý brook. Together with its own left-bank tributary it contains three ponds, whose size does not exceed 1 ha. Another one of the left-bank tributaries is the Malešický brook. On its 5 km long course the Temelínecký brook gains the altitude of approximately 73 m. The concerned populated area is the village of Malešice.

The key for these tables is identical with the key provided for table Tab. 83, Tab. 84 and Tab. 85.

Tab. 99 Profile characteristics of the "TEMELÍNECKÝ BROOK“ (Class IV)

watercourse:	Temelínecký potok	A	P_a	Q_a
hydrological order No.:	1-08-03-0792	[km^2]	[mm]	[m^3/s]
in profile:	upstream from the left-bank tributary from the "Pod Dubencem" location	2.09	617	0.007

Tab. 100 M-day flow rates of the "TEMELÍNECKÝ BROOK" (Class IV) watercourse

M	30	60	90	120	150	180	210	240	270	300	330	355	364
Q_M	19	12	9	7	5	4	3	2	2	1	0.9	0.4	0.2

Tab. 101 N-year flow rates of the "TEMELÍNECKÝ BROOK" (Class IV) watercourse

N	1	2	5	10	20	50	100	1000
Q_N	0.66	1.1	1.8	3.0	4.4	6.5	8.2	15.2

The **Sedlecký brook**, which is the main watercourse of the village of Temelín, contributes to the natural draining of the west side slope. The Sedlecký brook falls within the Bílý brook basin.

The **Bílý brook**, also called **Radomilický**, springs northeast from the village of Kaliště. For Temelín NPP it plays the role of recipient of the Temelínecký brook, which drains the southwest border of the area of interest of ETE3,4. The Bílý brook is an important local watercourse, with built pond complexes. For instance the Bělohůrecký pond, large complex of ponds near Dívčice or the cascade of Strpský and Mlýnský ponds. The most important tributaries include following brooks: Sedlecký, Temelínecký, Koutecký (with its complex of ponds), Dříteňský and Újezdecký. The concerned populated areas are the villages of Lhota pod Horami, Strachovice, Číčenice and Milenovice. The Bílý brook empties into the Blanice river as its left-bank tributary near the village of Milenice, approximately 13 km away from the area. The altitude difference on its approximately 22 km long course is about 116 m.

2.5.2.4.2 Local water surfaces

Standing water, which became a suitable place for biotope creation, was discovered on the unused premises of the former construction site of Temelín NPP. The standing water gradually turned into a wetland covering an area of approximately 5 ha. The **wetlands** create a suitable environment for many small vertebrates and amphibians, i.e. salamanders, newts, frogs and reptiles. Due to the biotope location in the area designed for the construction of ETE3,4 cooling towers, its recultivation is planned for 2012 to 2014. The area of interest will be drained. The collected water will be removed through the resulting sewer "A" in the current rain sewerage. Before rough levelling, the protected animal species will be transferred to selected substitute sites, see [L. 29].

Significant ponds, located on the area of interest of Temelín NPP, are situated in the southwest direction from the site. The ponds are mutually interconnected and create complexes that spread from Hluboká nad Vltavou up to Vodňany. The Bílý, also called Radomilický, brook and Bezdrevský brook are the main sources of water. The Radomilický brook falls within the Blanice basin; the Bezdrevský brook is a left-bank tributary of the Vltava river. The ponds are used for extensive fish farming. These include the following ponds:

Tab. 102 List of significant ponds on Radomilický and Bezdrevský brooks [L. 18]

name	Bezdrev	Munický	Plástovický	Vlhlavský	Blatec	Bělohůrský	Strpský
S [ha]	393.5	112.4	137.1	89.0	96.8	53.6	40.0
V [mil. m^3]	5627	843	726	1033	416	983	480

2.5.2.5 WITHDRAWAL OF PROCESS WATER FOR TEMELÍN NPP

The Vltava river, Hněvkovice reservoir, serves as the source of additional process water for ETE1,2. Study of possibilities for ensuring water withdrawal for ETE3,4 [L. 223], [L. 253], shows that increased water withdrawal is possible, if the entire storage volume of the Lipno I reservoir is used. The above cited study conservatively included the prediction of low water volume, which follows from the prognosis of possible climatic changes occurring in the 21st century.

Requirements on ensuring water withdrawals as of the reference year 2025 are realistic, including the preservation of a recreation surface and ensuring minimum flow rates below the reservoirs. A hypothetical operation of ETE1,2 and ETE3,4 was elaborated for the year 2085, considering extreme weather conditions at the maximum power plant performance. Comparison of monthly values of process water withdrawal shows that the highest amounts are withdrawn in July and August. The amount of water in summer months is lower in general. As for the Lipno I waterworks, the recreation surface level in summer months is currently maintained at 723.60 m above sea level. For the Hněvkovice reservoir the minimum surface level for summer months is 365.85 m above sea level. The prognosis for the reference year 2085, considering that the entire storage volume of the Lipno I reservoir is used to improve the Hněvkovice reservoir, points to an intervention in the manipulation regulations.

Tab. 103 Annual withdrawal values of additional process water and waste water discharge for conditions in 2025 [L. 223]

PERFORMANCE [MW_e]	WITHDRAWAL [m^3/year]	DISCHARGE [m^3/year]
2000	44,427,630	9,593,686
5200	108,835,600	23,448,770

The Temelín NPP pumping station of process water (ČSH) is located on the left river bank, near the Hněvkovice reservoir. Six pumping aggregates of the HWBW Sigma series are installed in the pumping station. Pumps in a monoblock arrangement have the specific energy parameter Y = 1717 ~ 1442 J/kg at the volume of pumped water of Q = 0.85 ~ 1.5 m^3/s. ČSH is constructed for the original planned output of the power plant, i.e. 4x 1000 MW and the aggregate therefore contains 1 ~ 4 pumps with 50 % reserve, for the proposed withdrawal of 1.3 ~ 4.16 m^3/s. Water is pumped to the 2x 15,000 m^3, water tank, which is located directly in Temelín NPP. With the current discharge parameters of 2xDN 1600, with the length of 6.20 km, it is possible to pump up to 4.8 m^3/s [L. 223].

Temelín NPP regularly analyses the withdrawal of raw water from the Hněvkovice reservoir. The amount of withdrawn water is determined indirectly; it is calculated from the energy consumption on raw water withdrawal in ČSH. Minimum residual flow underneath the Hněvkovice reservoir was determined in accordance with the regulation rules to be approximately 5.365 m^3/s [L. 225].

Withdrawal structure, ČSH and discharge pipes do not guarantee the water delivery from ČSH to Temelín NPP cisterns during the seismic load determined for designing systems, constructions and components relevant for nuclear safety. The ETE3,4 project will therefore, in accordance with note 11 on page 48 of IAEA NS-G-1.9 [L. 8] ensure enough water supply for this power plant site to put out of operation and cool down the blocks for at least 30 days.

Permission for surface water withdrawal

List of issued permissions for surface water withdrawal for the operation of ETE1,2 [L. 225]:

- Permission for water manipulation ref. No. 6804/93/Si of 15 December 1993 – point A). This permission became ineffective by the following decision of 27 February 2007.
- Decision ref. No. OŽP/7497/2006/Si of 27 February 2007 with a determined effective date until 31 December 2016

Tab. 104 Permitted water withdrawal for ETE1,2 valid until 2016 [L. 225]

Q_{MEAN} [l/s]	Q_{MAX} [l/s]	Q_{MONTH} [m^3/month]	Q_{YEAR} [m^3/year]
1,800	3,000	6,000,000	42,000,000

- Decision No. 08915/2010 of 14 June 2010 modifies the above cited decision on surface water withdrawal from a significant watercourse, the Vltava river, river km 210.46, construction site 66, cadastral office Litoradlice, čhp 1-06-03-076, HGR 632. Permission modification consists of an increased annual permitted amount, from 42 mil. m3/year to 47 mil. m3/year. The effective date of the manipulation permission changed from 31 December 2016 to 31 December 2026

2.5.2.6 WASTE WATER DISCHARGE

Discharge of waste water from ETE3,4 (for expected volumes see Tab. 103) is planned using the current waste distribution lines 2x DN700 from ETE1,2 to Kořensko [L. 222]. The necessity of possible sanitation of the current pipelines and of adding another line will be assessed based on requirements of the concrete project design of ETE3,4. Due to the fact that waste water discharge impacts the water quality of the recipient, the discharged amount and concentration is governed by the "Permission to discharge waste waters". In case the amount of waste increases, it is necessary to request a "Change of permission to discharge waste waters" in accordance with Act No. 254/2001 Coll. [L. 284].

Some of the pollution indicators for the Vltava and Lužnice rivers might improve soon due to meeting the C_{emis} standards in accordance with Government Decree No. 61/2003 Coll. [L. 286], for all pollution point sources and by stricter pollution limits for the water industry, pools and fish farming [L. 224].

Within a state task, Temelín NPP monitors the impact of discharged waste waters on the temperature increase and water quality in the Vltava river. The state task No. N 03-331-867 "Research of the impact of Temelín NPP on the hydrosphere and other components of the environment" is related to further use of the recipient as a source of drinking water after processing. Heat pollution of the Vltava river, as opposed to the annual water temperature in the river before the waste water discharge point, is +0.03°C. Heat pollution is very low.

Technological water and sanitary sewers

All technological waste water (in ETE1,2 it constitutes about 95 % of the waste water discharged from ETE1,2), together with cleaned sewer water are collected into the "Collection basin of technological water and sewers - 500 m^{3}". Sewer water is cleaned in the mechanical-biological waste water treatment plant, which is a part of ETE1,2. Technological water contains mainly the "blowdown" from the circulation cooling circuit and also waste water from the chemical treatment plant and low-activity technological water. The above mentioned "Collection basin of technological water and sewer water - 500 m^{3}" is a measuring object, which regularly monitors indicators such as BSK_5, $CHSK_{Mn}$, $CHSK_{Cr}$, sulphate, inorganic nitrogen, phosphate phosphorus, total phosphorus, suspended solids, non-polar extracted substances, anion tensides, RAS, pH and temperature in °C. An "Emergency plan" is approved for discharging in case of leak of hazardous substances into the Vltava river through the waste water. Waste water lines 2x DN 700 are used to discharge the waste water into the river. These lines are terminated in the "Termination and measurement building" on the left bank of the Vltava river. The hydroenergy potential of the lines is used through the Pelton turbine. Residual unused energy further improves the flow in the MVE suction cup, which is part of the Kořensko waterworks. Weir pillars, which evenly homogenize the water quality through the whole profile, are used to discharge waste waters in case the MVE is put out of service. Homogenisation prevents waste water stratification in the Orlík reservoir. Indicators such as Ph, temperature, TOC (organic substances, dissolved oxygen and conductivity, are monitored at the waste water discharge point in the Kořensko profile.

Currently the $CHSK_{Cr}$ indicators reach critical values in the recipient. Annual average values of $CHSK_{Cr}$ are only a little lower than the derived average standard in accordance with the Government Decree No. 61/2003 Coll. of 29 January 2003 on indicators and values for admissible levels of surface and waste water pollution, on formalities concerning permission to release sewage into surface waters and into the sewer system, and on sensitive areas. Values of the discharged $CHSK_{Cr}$ pollution depend on the quality of withdrawn raw water [L. 224].

Low-activity technological waters

Samples of so-called over-balance technological tritium water are taken from tanks in the controlled area of ETE1,2. The volume activity of samples is determined by the spectrometric or radiochemical analysis and its value is compared to discharge limits. Sampling frequency and methods of determining the amount of radioactive substances are governed by the "Temelín NPP monitoring plan". Technological waste waters are subsequently moved to the "Collection basin of technological water and sewer water – 500 m^3".

A prognosis of average concentration of discharged radioactive substances has been processed for the current operation of ETE1,2 and ETE3,4. Impacts of radiotoxic effect of discharged radioactive substances on water biocenosis caused by the operation of ETE1,2 have not been determined. The forecast levels of volume activities of radioactive substances (tritium, strontium, caesium and other AAFP) during the operation of ETE1,2 and ETE3,4 are significantly lower than the derived pollution limit. A classic representative of severe impact on hydrosphere, from the mixture of other AAFP, is radionuclide Ceasium 137. The forecast levels of Ceasium volume activities are significantly lower than the derived pollution limit, similarly to tritium. Detected impact by remote transmission through water beyond the borders of the Czech Republic applies only to Tritium [L. 224]. Tritium volume activities in the Elbe - Hřensko profile are significantly lower than the stated indicative value for drinking water in accordance with EU Directive 98/83/EC [L. 228].

GROUND WATERS

Temelín NPP already observes 49 monitoring wells in the power plant site and its surroundings. In addition, 51 drain wells were established. Water quality and the ground water regime is assessed. The results are available to the public in the Temelín NPP information centre in the form of annual reports for comparison. The effect of Temelín NPP operation on ground waters is clear from Section 2.8.2.13 of this report.

Permission for water use

List of valid water use permissions to discharge waste water:

- Permission for water manipulation ref. No. 6804/93/Si of 15 December 1993 – point B) valid for the duration of trial and regular operation
- Decision ref. No. 10424/93/01-231/2-Si of 8 March 2002, amending the permission for water manipulation ref. No. 6804/93/Si of 15 December 1993 – valid for the duration of trial and regular operation.
- Decision ref. No. KUJCK 10012/2004 OZZL-Ža of 14 April 2004, amending decision 10424/93/01-231/2-Si of 8 March 2002 in the "suspended solids" indicator with effect until 31 December 2013.

- Decision ref. No. KUJCK 18378/20/2005 OZZL-Ža of 22 January 2007, statement related to discharging radioactivity indicators, amending Decision ref. No. 6804/93/Si of 15 December 1993, part B II – Radioactivity indicator values stated on page 4 of this decision (ref. No. 6804/93/Si) and also cancels conditions of this decision stipulated under No. 2,3,4,5,6,7 and 8 – valid for the duration of trial and regular operation (see decision ref. No. 6804/93/Si).
- Decision ref. No. SÚJB/OROPC/26161/2009 of 1 December 2009, permission for introducing radionuclides into the environment in the form of liquid discharges - permission is valid for an indefinite period of time

Permission to discharge waste water ref. No. 6804/93/Si of 15 December 1993, into the recipient, the Vltava river, at 200.405 river km, as stated in the cadastral office of Neznašov; the permission is valid for the duration of trial and regular operation. Should Temelín NPP need to request a new permission to discharge waste waters in the Kořensko profile, the Regional Government shall set the term of effect to 4 years at most. In case the amount of waste increases, it is possible to request the "Change of permission to discharge waste waters" issued by the relevant Water Management Office.

2.5.2.7 WATERWAY

The Vltava river is currently navigable from river km 0.0 - i.e. the mouth into the Elbe river near Mělník up to river km 91.5 - Třebenice downstream from the Slapy waterworks. Watercourse parameters in this section fall within Class IV in accordance with the international classification. Operation of ETE1,2 and its extension for ETE3,4 does not have a negative effect on the waterway traffic on the Vltava river [L. 223].

Construction of the waterway on Horní Vltava (from river km 91.5 - Třebenice up to river km 239.6 - České Budějovice) was included in the transport infrastructure construction schedule between 2008 and 2013, which was approved by the Government of the Czech Republic in September 2007. This section is considered a significant transport waterway for ships with bearing capacity up to 300 t. Parameters of the waterway section meet the Class I requirements, in accordance with the international classification of waterways.

Currently the entire storage volume is considered a reservoir of technological water for ETE1,2. The expected increase in current withdrawal volumes for ETE1,2 on average is from Q = 1.1 ~ 1.5 m^3/s to Q = 3.1 ~ 3.8 m^3/s. However, current volumes are withdrawn with practically constant operation surface level and do not effect the navigable depth. The approved scope of storage volume surface level is 5.50 m.

2.5.3 REQUIREMENTS AND CRITERIA

Tab. 105 Assessment criteria for hydrological risks

ID	Article	Criteria requirement in Decree No. 215/1997 Coll.	Par.	Requirements in IAEA NS-R-3
6.1	**Article 4p)**	Intersection of the site with flood areas of watercourses which are flooded at Q_{100}[23] or on areas that might be flooded as a consequence of a water reservoir/dam accident.	3.18 3.19. 3.20. 3.21 3.22. 3.23. 3.29. 3.30. 3.31.	**Floods due to precipitation and other causes** The region shall be assessed to determine the potential for flooding due to one or more natural causes such as runoff resulting from precipitation or snow melt, high tide, storm surge, seiche and wind waves that may affect the safety of the nuclear installation. If there is a potential for flooding, then all pertinent data, including historical data, both meteorological and hydrological,shall be collected and critically examined. A suitable meteorological and hydrological model shall be developed with account taken of the limits on the accuracy and quantity of the data, the length of the historical period over which the data were accumulated, and all known past changes in relevant characteristics of the region. The possible combinations of the effects of several causes shall be examined. For example, for coastal sites and sites on estuaries, the potential for flooding by a combination of high tide, wind effects on bodies of water and wave actions, such as those due to cyclones, shall be assessed and taken into account in the hazard model. The hazards for the site due to flooding shall be derived from the model. The parameters used to characterize the hazards due to flooding shall include the height of the water, the height and period of the waves (if relevant), the warning time for the flood, the duration of the flood and the flow conditions. The potential for instability of the coastal area or river channel due to erosion or sedimentation shall be investigated. **Floods and waves caused by failure of**

[23] In accordance with ČSN 75 1400 Hydrological data on surface waters, 1990.

ID	Article	Criteria requirement in Decree No. 215/1997 Coll.	Par.	Requirements in IAEA NS-R-3
				water control structures Information relating to upstream water control structures shall be analysed to determine whether the nuclear installation would be able to withstand the effects resulting from the failure of one or more of the upstream structures. If the nuclear installation could safely withstand all the effects of the massive failure of one or more of the upstream structures, then the structures need be examined no further in this regard. If a preliminary examination of the nuclear installation indicates that it might not be able to withstand safely all the effects of the massive failure of one or more of the upstream structures, then the hazards associated with the nuclear installation shall be assessed with the inclusion of all such effects; otherwise such upstream structures shall be analysed by means of methods equivalent to those used in determining the hazards associated with the nuclear installation to show that the structures could survive the event concerned.
6.2			3.32.	The possibility of storage of water as a result of the temporary blockage of rivers upstream or downstream (e.g. caused by landslides or ice) so as to cause flooding and associated phenomena at the proposed site shall be examined.
6.3			4.4.	A description of the surface hydrological characteristics of the region shall be developed, including descriptions of the main characteristics of water bodies, both natural and artificial, the major structures for water control, the locations of water intake structures and information on water use in the region.

ID	Article	Criteria requirement in Decree No. 215/1997 Coll.	Par.	Requirements in IAEA NS-R-3
6.4			3.53	In the design of systems for long term heat removal from the core, site related parameters, such as the following, should be considered: (a) Air temperature and humidity; (b) Water temperatures; (c) Available flow of water, minimum water level and the period of time for which safety related sources of cooling water are at a minimum level, with account taken of the potential for failure of water control structures.
			3.54	Potential natural and human induced events that could cause a loss of function of systems required for the long term removal of heat from the core shall be identified, such as the blockage or diversion of a river, the depletion of a reservoir, an excessive amount of marine organisms, the blockage of a reservoir or cooling tower by freezing or the formation of ice, ship collisions, oil spills and fires. If the probabilities and consequences of such events cannot be reduced to acceptable levels, then the hazards for the nuclear installation associated with such events shall be established.
6.5			3.24	The region shall be evaluated to determine the potential for tsunamis or seiches that could affect the safety of a nuclear installation on the site.
			3.25	If there is found to be such a potential, prehistorical and historical data relating to tsunamis or seiches affecting the shore region around the site shall be collected and critically evaluated for their relevance to the evaluation of the site and their reliability.
			3.26	On the basis of the available prehistorical and historical data for the region and comparison with similar regions that have been well studied with regard to these phenomena, the frequency of occurrence, magnitude and height of regional tsunamis or seiches shall be estimated and shall be used in determining the hazards associated with tsunamis or seiches, with account taken of any amplification due to the coastal configuration at the site.

ID	Article	Criteria requirement in Decree No. 215/1997 Coll.	Par.	Requirements in IAEA NS-R-3
			3.27	The potential for tsunamis or seiches to be generated by regional offshore seismic events shall be evaluated on the basis of known seismic records and seismotectonic characteristics.
			3.28	The hazards associated with tsunamis or seiches shall be derived from known seismic records and seismotectonic characteristics as well as from physical and/or analytical modelling. These include potential draw-down and runup4 that may result in physical effects on the site.

2.5.4 DOCUMENTS PROVIDING A BASIS FOR THE ASSESSMENT

- Povodí Vltavy, state-owned enterprise www.pvl.cz [L. 277]
- Prof. Ing. František Hrádek, DrSc.: Posouzení účinků extrémních srážek s dobou opakování N = 100, 1000 a 10 000 let na povrchový odtok v areálu JE Temelín (Assessment of Extreme Precipitation Effects with a Period of Repetition of N = 100, 1000 and 10,000 Years on Surface Discharge on the Premises of Temelín NPP) – Hydrological Study, 1998 [L. 221]
- L. Assessment of the Waste Water Discharge System from Temelín NPP to Kořensko ÚJV Řež a.s., Division ENERGOPROJEKT, Prague 02/2008 [L. 222]
- Ing. Kašpárek: Studie možnosti zajištění odběrů vody z VD Hněvkovice pro výhledové rozšíření JE Temelín (Feasibility study of water withdrawal from the Hněvkovice reservoir for the future extension of Temelín NPP), T. G. Masaryk Water Research Institute Prague, 2009 [L. 223]
- Ing. Hanslík: Podpůrná studie EIA – Nový jaderná zdroj v lokalitě ETE - posouzení vlivu na povrchové vody (Support EIA Study – New Nuclear Installation at Temelín NPP - Surface Water Impact Assessment); T. G. Masaryk Water Research Institute, 2009 [L. 224]
- Předprovozní bezpečnostní zpráva ETE 1,2 (Pre-operation safety report of ETE1,2) [L. 18]
- Updated hydrological data for surface flows in the area of Temelín NPP, Czech Hydrometeorological Institute České Budějovice, 16 February 2011 [L. 226]
- SÚJB BN-JB-1.14, Interpretace kritérií pro umísťování jaderných zařízení a návrh jejich průkazů (Interpretation of criteria for the siting of nuclear installations and a proposal for their evidence documentation), SÚJB, April 2012 [L. 268]

Permission for water use

- Permission for water handling ref. No. 6804/93/Si of 15 December 1993 [L. 225]
- Decision ref. No. 10424/93/01-231/2-Si of 8 March 2002, amending the permission for water handling ref. No. 6804/93/Si of 15 December 1993 [L. 225]
- Decision ref. No. KUJCK 10012/2004 OZZL-Ža of 14 April 2004, terminating the permission from 2002 [L. 225]
- Decision ref. No. KUJCK 18378/20/2005 OZZL-Ža of 22 January 2007, statement relating to discharge of radioactive indicators [L. 225]
- Decision ref. No. OŽP/7497/2006/Si of 27 February 2007 concerning the permission for surface water withdrawal, with a definite period of validity by 31 December 2016 [L. 225]
- Decision ref. No. SÚJB/OROPC/26161/2009 of 1 December 2009, permission for introducing radionuclides into the environment in the form of liquid discharges - permission is valid for an indefinite period of time [L. 225]
- Decision ref. No. 08915/2010 (OŽP/08915/2010/F) of 14 June 2010, change of the permitted quantity of annual consumption of surface water, validity by 31 December 2026 [L. 225]

2.5.5 METHODS APPLIED TO THE EVALUATION

Measurement and evaluation of hydrological data of surface waters

Hydrological measurements are performed by standard methods in the network water measuring stations with a continuous level recording and conversion to flow rate through current stage-discharge curves [L. 226]. Basic hydrological data are processed from the measurement results in accordance with ČSN 75 1400.

Hydrological data for the initial (reference) safety report, which are beyond the observed profiles, were processed by the Czech Hydrometeorological Institute, using the so-called discharge inventories (see [L. 226]). The principle of the inventory is to distribute the results from the stations into the network of adjacent profiles and unobserved tributaries. The basis for the calculation, in addition to the comparison results of station observations, is so-called regional regression analysis. The analysis yields regression equations, which express the relationship between the flow rate and geographical characteristics of the basin. These equations enable us to calculate the relative distribution of theoretical probability distribution parameters among basins with unobserved tributaries based on their different geographical characteristics. The relative distribution is then balanced using the iteration procedure to obtain the minimum deviation against the observed results in the stations after the balance sum is calculated. The discharge inventories represents a full-scale hydrological regime model for the areas of interest, verified by the observed results. The observation period for the inventory calculation for M-day and N-year flow rates differs. Standardized periods between 1931 – 1960, 1931 – 1980 and the latest, as yet unprocessed period between 1981 – 2010 are used for average long-term flow rates Q_a and M-day flow rate Q_{Md}. The always available maximum is used for N-year flow rates, i.e. from the starting records up to the present of the processed period.

Strouha, Palečkův brook and Temelínecký brook

Measured average flow rate Q_a and M-day flow rates Q_{Md} for Strouha, Palečkův and Temelínecký brook were processed [L. 226] in relation to the discharge inventory Q_{Md} for the 50-year period between 1931 – 1980, with the theoretical probability distribution LP 3. The discharge inventory for 1981 - 2010 is currently being processed; completion of this process is planned for the end of 2012. No significant change is expected in the new inventory for the above stated brooks.

N-year flow rate values were processed in accordance with the new discharge inventory Q_N, which was compiled based on observation from its beginning up to 2003. Data on Q_{1000} were determined using extrapolation from theoretical probability distribution parameters that are included in the discharge inventory ($Q_{max,mean}$, C_v, C_s, distribution LN 3). Q_{1000} does not belong to basic data in accordance with ČSN 75 1400 [L. 226].

Vltava – Hněvkovice and Kořensko

The Hněvkovice and Kořensko waterworks profiles are located on the observed section of the river. Data could therefore be processed based on observations from the last decade.

M-day flow rates for the period between 2001 - 2010 were processed from the observations on stations in České Budějovice on the Vltava river and in Bechyně on the Lužnice river [L. 226]. Conversion into the required two profiles was performed using the same methodology as for the discharge inventory. These values are affected by the operation of waterworks located in the basin upstream from the profile, namely the Lipno I and Lipno II waterworks complex.

N-year flow rates are a result of new discharge inventory of N-year flow rates processed after the occurrence of extreme floods in August 2002. The processed period contains maximum available data series from the second half of the 19th century up to the year 2003. The nearest relevant hydrometric station, on which the discharge inventory for the Hněvkovice and Kořensko waterworks is based, are located in České Budějovice on the Vltava river and in Bechyně on the Lužnice river. In addition, observed flow rate data series from already closed stations in Hluboká and Týn on the Vltava river, were also used. For safety reasons the data are cleared of the influence of large reservoirs. These data therefore correspond to the natural post-flood regime without the influence of the Lipno reservoir [L. 226].

Data on Q_{1000} (do not rank among basic data according to ČSN 75 1400) were determined using extrapolation from theoretical probability distribution parameters that are included in the discharge inventory ($Q_{max,mean}$, Cv, Cs, distribution LN 3). Data on Q_{1000} is so distant from basic hydrological data according to ČSN 75 1400 that a special study needed to be prepared. This study was carried out on the central division of surface waters in the Czech Hydrometeorological Institute in Prague specially for the Hněvkovice waterworks only [L. 226].

2.5.6 DEFINITION OF THE AREA EXAMINED

As far as hydrology is concerned, the NPP locality is located on the divide of two rivers, the Vltava and Bílý brook. The Bílý brook is a right-bank tributary to the Blanice river, which empties into the Vltava river. The examined area falls within the Vltava basin.

2.5.7 DETAILED ASSESSMENT OF ALL REQUIREMENTS AND CRITERIA DEFINED IN DECREE NO. 215/1997 COLL., IN COMBINATION WITH IAEA NS-R-3

2.5.7.1 CRITERION BASED ON ITEM 6.1

2.5.7.1.1 Criterion defined by Article 4 par. p) of Decree No. 215/1997 Coll.

2.5.7.1.1.1 Reference to the text of the criterion

The text of the criterion defined in Article 4 par. p) of Decree No. 215/1997 Coll. [L. 1], is reproduced in Item 6.1 in Tab. 105.

2.5.7.1.1.2 Possibility of flooding of the ETE3,4 site from local watercourses

The power plant site is located on the divide (see Section 2.5.2.1 of this report) in the altitude ranging between 503.00 ~ 507.00 m above sea level (see Section 2.1.2.2.1 of this report). Nuclear safety-related buildings are located 507.00 m above sea level. Smaller watercourses in the close proximity of the Temelín NPP site (see attached drawing Dwg. 2) spring at the altitude of approximately 490.00 m above sea level. Minimum elevation of the ETE3,4 site above these springs is at least 10 m. These watercourses mainly consist of torrents with a steep fall and maximum flow rates at the mouth to the Vltava river Q_{100} = 16 m^3/s. There are no water reservoirs on these rivers that could flood the Temelín NPP surroundings in case of an accident.

As follows from the above mentioned, the ETE3,4 site is not at risk from floods from surrounding watercourses.

2.5.7.1.1.3 Possibility of flooding of ETE3,4 from the Vltava river

Hněvkovice and Orlík are water reservoirs situated nearest to the power plant located in the area of interest of Temelín NPP. The maximum surface level in the Hněvkovice reservoir during flow regulation in the Vltava river is 370.10 m above sea level. In the Orlík reservoir, which is situated at a lower altitude, the maximum surface level during flow regulations reached 353.60 m above sea level (see Tab. 106).

Tab. 106 Surface levels in reservoirs on the Vltava river

Water surface	Flow regulation		Break wave in Lipno I	
	H_{MAX} [m above sea level]	Elevation [m]	$H_{Q10\,000}$ [m above sea level]	Elevation [m]
Hněvkovice	370.10	7.65	376.70	14.25
Orlík (Kořensko weir)	353.60	3.00	364.90	14.30

The close vicinity of the Vltava river contains the water management buildings of Temelín NPP pumping station (ČSH) and the "Building for valves and measurement", which are required for the normal operation of Temelín NPP. The maximum surface level in adjacent water reservoirs range between 3.00~7.65 m below the entry levels to the 1st above-ground floor of given buildings (for comparison see Tab. 106 and Tab. 107). The operation of Temelín NPP located on the bank of the Vltava river is not at risk during flow regulation on the Vltava due to the above stated elevation of Temelín NPP objects (see column "flow regulation" → "elevation" in Tab. 106).

Tab. 107 Dimensions of Temelín NPP objects situated on bank of the Vltava river

Building	Corresponding water surface from table Tab. 106	1.NP±0.00 m [m above sea level]
Raw water pumping station	Hněvkovice	378.18
Building for valves and measurement	Kořensko	357.30

Floods caused by a water reservoir accident. The effect of a break wave in the Lipno I reservoir on the Hněvkovice reservoir was simulated using the model calculation [L. 226]. The breaking of a 25 m high dam of the Lipno I reservoir with a crack 22 m (at the abutment) and 46 m (at the overflow level) wide was supposed for the conversion of $Q_{10,000}$. The waterworks is located approximately 115 km from Temelín NPP. At culmination of $Q_{10,000}$ the surface level at the dam of the Hněvkovice reservoir would reach 376.60 m above sea level and 364.90 m above sea level at the Kořensko weir. On average the values are higher than the normal surface level for 14 m. The raw water surface station and the Building for valves and measurement will be flooded and put out of operation, which will result in ETE3,4 being put out of service. The flow of simulated flood wave from Lipno to the Hněvkovice waterworks creates a time delay in which ETE3,4 blocks can be put out of operation and cooled down[24].

The elevation of Temelín NPP site (as already stated, located at an altitude ranging between 503.00 ~ 507.00 m above sea level) over the maximum surface level in reservoirs on the Vltava river is at least 135 m. The Temelín power plant site itself is therefore not at risk from even hypothetical floods from the main recipient in the area.

Centennial flood water Q_{100} (see the situation in 2002 and [L. 226]) will not effect the nuclear safety of ETE3,4. Flooding of ČSH and the discharge building at the Hněvkovice weir will put blocks ETE3,4 out of service and does not impact nuclear safety.

The Temelín NPP locality meets requirements stipulated in Article 4 par. p) of Decree No. 215/1997 Coll. [L. 1].

2.5.7.1.2 Requirements in Section 3.18 of IAEA NS-R-3

The text of the requirement in Section 3.18 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 105 under item 6.1.

Consequences of putting out of service of the rain sewerage and the whole drainage system were analysed by the study of extreme precipitation effect on the surface drain from the Temelín NPP site [L. 221]. The results are expressed in water accumulation values for the given surface at one-day precipitation. Average accumulation value with a repetition time of N = 100 years is 44.3 mm; with a repetition time of N = 10,000 this is 84 mm. The above cited analysis proved that buildings, their entries, floors and important equipment are located above the flood level or are equipped with watertight covers. Therefore the flood does not pose any

[24] Storage of water for compensating for losses in cooling circuits in ETE1,2 is ensured namely in the additional water reservoir 2x 15,000 m^3, in the tertiary circuit (cooling channels and cooling pools) and, if necessary, the circuit can be filled from the drinking water distribution. Ensuring reservoirs of drinking water for cooling down ETE3,4 is a part of its design requirements.

significant risk to the power plant site; the rain water will be drained outside the area of interest by the surface drain.

The ETE3,4 site meets the requirements in Section 3.18 of IAEA NS-R-3 [L. 6].

2.5.7.1.3 Requirements in Section 3.19 to 3.23 of IAEA NS-R-3

The text of the requirements in Sections 3.19 to 3.23 of IAEA NS-R-3 [L. 6] is reproduced under item 6.1 in Tab. 105.

These requirements apply to the methodology of measures and data evaluation for floods that were met by the procedure used by the Czech Hydrometeorological Institute to process the documents providing basis for this report (see [L. 226]).

Requirements in Section 3.20 are related to the power plant construction site on a seaside or near the coastline and therefore do not apply to the Temelín construction site.

Dredging of the Vltava river channels, performed by the basin manager, removes the sediment layer and ensures the long-term stability of the recipient channel supplying additional process water to ETE3,4.

The ETE3,4 site meets the requirements in Sections 3.19 and 3.23 of IAEA NS-R-3 [L. 6], where applicable to inland localities.

2.5.7.1.4 Requirements in Section 3.29 to 3.31 of IAEA NS-R-3

The text of the requirements in Sections 3.29 to 3.31 of IAEA NS-R-3 [L. 6] is reproduced under item 6.1 in Tab. 105.

Requirements on the methodology of data evaluation for floods caused by dam defect on the upper flow of the river, meeting the criterion stipulated in Article 4 par. p) of Decree No. 215/1997 Coll. [L. 1], which was analysed in Section 2.5.7.1.1.3 of this report. The required procedure was fulfilled by the methodology used by the Czech Hydrometeorological procedure to process documents providing a basis for this report [L. 226].

Flood resulting from a break in the Lipno I reservoir dam above the withdrawing building and process water pumping station for Temelín NPP will result in flooding of the process water pumping station and putting it out of operation. However, the production buildings and nuclear safety-related buildings will not be flooded, as they are located approximately 127 m above the bank with the pumping station affected by hypothetical flood level ($Q_{10\ 0000}$).

The design basis for Temelín NPP performance management follows from the described event: in case of a flood wave that will put the additional water pumping station out of operation, all operated Temelín NPP blocks will be put out of operation and then cooled down.

Ensuring water supply for putting out of operation and cooling down floods is also an issue related to the flooding and putting out of operation of ČSH, see Section 2.5.7.2.

The ETE3,4 site location meets the requirements stipulated in Sections 3.29 to 3.31 of IAEA NS-R-3 [L. 6].

2.5.7.2 REQUIREMENTS IN SECTION 3.32 OF IAEA NS-R-3

The text of the requirement in Section 3.32 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 105 under item 6.2.

Additional process water for the operation of Temelín NPP is withdrawn from the retention volume of the Hněvkovice waterworks (for capacity see Tab. 82) with sufficient capacity for at least one month of the power plant operation at the nominal output. The hypothetical possibility of blocking the water intake from the upper section of the Vltava river flow will not affect the supply of this water for ETE3,4.

Additional water supply into ETE3,4 can be interrupted, however, by flooding of the ČSH, power outage in ČSH or putting out of operation of water delivery pipes from ČSH to water reservoirs in the Temelín NPP site. Therefore a supply of water sufficient for removing the residual heat from the active core of blocks 3 and 4 will be ensured for at least 30 days after putting out of operation. Ensuring reservoirs of drinking water for putting out of operation and cooling down ETE3,4 is a part of its design requirements listed in Section 2.10 of this report.

The ETE3,4 site location meets the requirements stipulated in Sections 3.29 to 3.31 of IAEA NS-R-3 [L. 6].

2.5.7.3 REQUIREMENTS IN SECTION 4.4 OF IAEA NS-R-3

The text of the requirement in Section 4.4 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 105 under item 6.3.

Description of hydrological characteristics of the area, including the main characteristics of water buildings and their use in accordance with Section 4.4 of IAEA NS-R-3 [L. 6], is summarized in Section 2.5.2 of this report.

2.5.7.4 REQUIREMENTS IN SECTION 3.53 AND 3.54 OF IAEA NS-R-3

The text of the requirement in Section 3.53 to 3.54 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 105 under item 6.4.

The possibility of an outage of the additional process water supply from the ČSH place of withdrawal to the heat removal system of the active core cannot be ruled out; the ETE3,4 project will need to include a technical solution to ensure the water supply for the heat removal system of the active core so as to ensure its functionality after blocks ETE3,4 are put out of operation and cooled down. This design requirement has already been stated in Section 2.5.7.2 of this report.

2.5.7.5 REQUIREMENT IN SECTIONS 3.24 TO 3.28 OF THE IAEA NS-R-3 STANDARD

The text of the requirement in Sections 3.24 to 3.28 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 105 under item 6.5.

Tsunami can occur on open sea or ocean during a strong earthquake, deep-sea earth slide or by meteor impact. Tsunami refers to several subsequent waves that can reach up to cm in height and 700km/h in speed on the open and deep sea. When they reach the coast, due to friction against the sea bottom, their speed decreases and they rise up to several tens of metres. Tsunami behaviour is governed by the wave length.

There are no historical records of tsunami in the given geographical and climatic conditions of Central Europe, nor are there conditions for tsunami occurrence.

Seiche-type waves have not been observed in the Czech Republic. There is no sufficiently large water surface near Temelín NPP that could cause seiche waves as result of an earthquake.

Requirements stipulated in Sections 3.24 to 3.28 of IAEA NS-R-3 [L. 6] do not apply to the Temelín NPP locality.

2.5.8 CONCLUDING ASSESSMENT

The following conclusions can be made based on the hydrological background data and their assessment in accordance with applicable criteria and requirements:

- With regards to the flood risk of lands with buildings, systems and components related to nuclear safety, no conflict with the criterion in accordance with Article 4 par. p) of Decree No. 215/1997 Coll. was discovered [L. 1] The ETE3,4 site is not at risk of being flooded by local watercourses (see Section 2.5.7.1.1.1 of this report), nor by a flood situation at the Vltava river (see Section 2.5.7.1.1.3 of this report).
- Based on the assessment of the ETE3,4 site and water supply system for additional process water in accordance with requirements stipulated in Sections 3.18 to 3.23, 3.29 to 3.32 and 4.4, periodicity of flood as the initiating event was determined at N=10,000 years (see Sections 2.5.7.1.1.3 and 2.5.7.1.4 of this report). This flood will flood ČSH and Temelín NPP discharge building and will put them out of operation, which will require putting out of operation and cooling down of Temelín NPP blocks. Temelín NPP blocks will remain in this state until the operation of ČSH and the wastewater discharge building is resumed. In addition to putting blocks out of operation, the design measures also include ensuring the required supply of water to put out of operation and cool down the power plant block for a period of 30 days or longer, if the conservative analysis of the design does not allow a shorter time to be substantiated[25].

Design requirements of the corresponding event and requirements described in point 2 of the previous paragraph is defined in Section 2.10 of this report.

[25] The requirement for water supply in the ETE3,4 site for a period of 30 days or longer is stipulated in section 4.141 of IAEA NS-G-1.9 [L. 8].

2.6 GEOLOGICAL, GEOTECHNICAL AND SEISMIC CONDITIONS

2.6.1 SCOPE OF THIS SECTION

The subject of this section is to summarize the available information and data from surveys of the geological, geotechnical, seismological and hydrogeological conditions of the near region of Temelín and their comparison with the criteria and requirements for siting of nuclear installation.

2.6.2 RECAPITULATION OF FINDINGS

2.6.2.1 BRIEF CHARACTERISTICS OF THE TEMELÍN REGION

2.6.2.1.1 Geological and Tectonic Structure of the Region

The region of Temelín NPP extends to two basic units of the geological structure of Europe – i.e. the Hercynides and Alpides. The near region of Temelín NPP lies in the Bohemian Massif (BM), which is a part of the European Hercynian orogeny[26]. The European Hercynides are a remnant of the wide, folded mobile zone wedged in between the East European or Fennosarmatian platform (EEC)[27], bordering the epi-Caledonian platform in the west, and the northern edge of the European Alpides (see also lit. [L. 138], [L. 108]). The inner structure of the Central European Hercynides is shown in Fig. 18.

As a part of the Moldanubian Zone of the Hercynides, the Bohemian Massif may be characterised as a polyphase formed geological unit which, despite being largely determined by elements of Hercynian age, also bears features of older geotectonic cycles, namely Caledonian[28], Cadomian[29] and pre-Cadomian. According to the generally accepted theory, the pre-platform stage of development of the Bohemian Massif ended in the orogenic phase of the Hercynian geotectonic cycle in the late Paleozoic (for details see basic material [L. 138]).

The Alpides are formed by the Alps stretching south of the Bohemian Massif and the Carpathian Mountains lying in the southeast and east (see Fig. 19). The subsurface contact of the Alpides with their platform foreland is planar. It is represented by a flat tectonic slide of nappes and blocks of the Alps and the Carpathians onto the southern edge of the platform. Based on geological and geophysical evidence confirmed by deep drills, the Hercynian platform submerged under the Alps extends to a distance of 30 - 40 km from the front of the Alpides. The surface border is associated with the range of the folded formations of the Alpine-Carpathian system Alpine-Carpathian Foredeep. It extends from Genoa via the arc of the Swiss Alps between Bern and Zurich, from there it continues along the Danube River in Austria and further towards Znojmo and Ostrava, it bends southwards below Krakow and

[26] Hercynian geotectonic cycle: age 365 – 255 Ma (formations: from Devonian to Permian).

[27] Platform – A stabilized portion of the Earth's crust with less intense tectonic activity and smaller motion gradient.

[28] Caledonian geotectonic cycle: age 575 – 390 Ma (formations: from Cambrian to Silurian).

[29] Cadomian geotectonic cycle: age 650 – 600 Ma (formation: Vendian).

continues along the Carpathian Mountains where it ends in the arch of the Eastern Carpathians by the Danube.

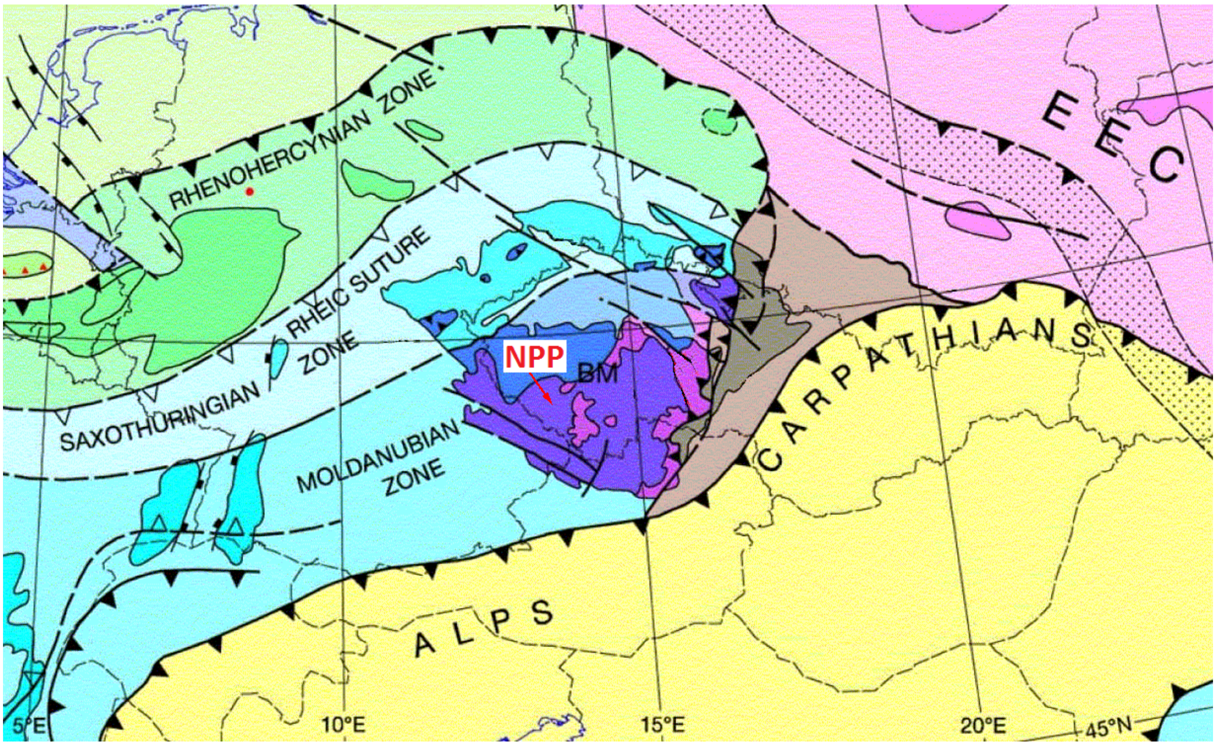


Fig. 18 Basic structural diagram of Central Europe. Taken from the TESZ project presentation (see basic material [L. 154]).

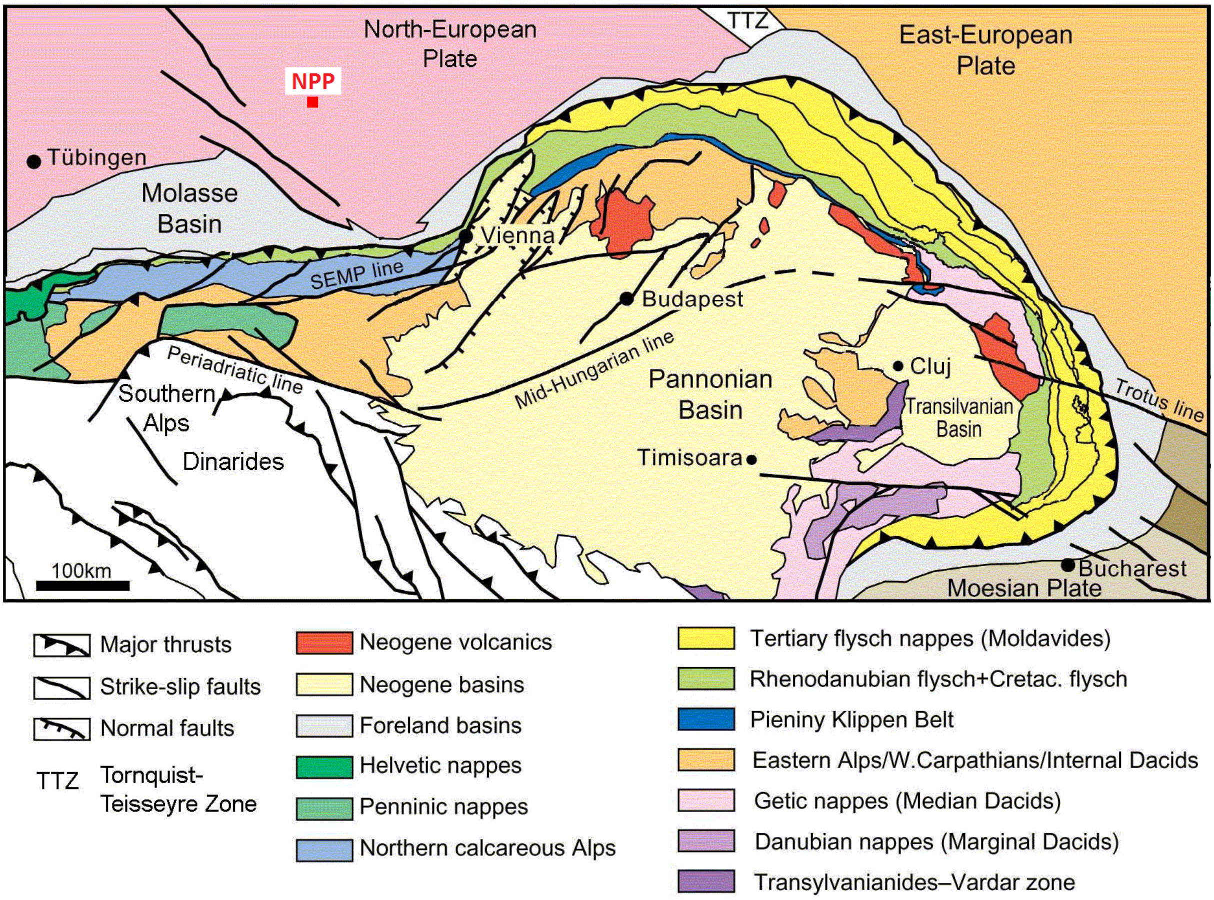


Fig. 19 Basic structural units of the Eastern Alps and Carpathians. Excerpt from the materials of the PANCARDI project (The Pannonian Basin, Carpathian arc, Dinaride) - (see basic material [L. 143]).

2.6.2.1.2 Seismicity of the Region

Apart from the geological structure and the age of the orogenic processes, the above described basic units of the geological structure of the Temelín NPP region are markedly distinguished by the rate, frequency and character of seismic activity. In terms of seismicity, the Alpides are a unit with disproportionately higher activity, although several areas may be found in the Hercynides, which require certain attention in relation to the occurrence of earthquakes.

Strong earthquakes in the region of Temelín NPP are causally associated with faults of all types, i.e. normal faults, strike-slip faults, and thrusts. The types as well as other geological properties of significant faults in the region are described and interpreted in a number of expert publications. For the purpose of seismic evaluation, the data were sorted, modified and subsequently utilised for a deterministic (seismotectonic) estimation of SL-2 (see basic material [L. 187], [L. 171] and [L. 18]). The databank of findings on the seismogenic and seismotectonic structures, as well as other geological and geophysical properties of the Temelín NPP region [see L. 160] is continuously complemented with newly published data.

Maps of the foci of historical earthquakes, plotted with the aid of national or compiled catalogues, unambiguously indicate that the seismic activity of the Temelín NPP region is concentrated mainly in its southern and southeastern part (see Fig. 20). Here, source areas of earthquakes may be found that have generated (or may generate) the largest ground motions in the near region of Temelín NPP.

Probably the most significant, seismically active structure in the region, in terms of generated ground motion, is the northwestern border of the ALCAPA block, represented by the Mur-Mürz-Leitha line and the Peripieniny lineament. These are tectonic structures formed in the Miocene, under N-S and NW-SE paleostress field conditions [L. 79]. Since the Miocene, the tectonic activity of the system has been determined by the pushing out of the ALCAPA megablock and its moving (sinistral strike-slip) from the Alpine Collision Zone eastwards. Recent GPS measurements and ALCAPA block geometry models have shown that the velocity of the left slip along the marginal faults of the Vienna Basin reaches 1-2 mm per year (see e.g. [L. 86]). The seismogenic area along the Mur-Mürz-Leitha line manifests itself in the earthquake focus maps as a clearly defined, narrow linear belt that stretches from Judenburg across the valleys of the Mur and the Mürz rivers and continues along the southeastern margin of the Vienna Basin up to the southern edge of the Little Carpathians. From there, it extends to the NE along the Peripieniny lineament. The macroseismic magnitudes of historical earthquakes along this line reached almost M = 6.1 (Katschberg, 1201 – see [L. 126]).

Another major seismogenic structure is associated with the faulting accompanying the subsurface contact of the Bohemian Massif and the Eastern Alps where foci of strong earthquakes have appeared (e.g. Riederberg, 1590, M = 5.8 - see [L. 126]).

In the south, the Temelín NPP region partially extends to another significant seismogenic area in northern Italy and western Slovenia (i.e. the Friuli – Bovec – Idrija source area) It is a very mobile area with marked horizontal movement of the Adriatic block towards the north (at a velocity from 1 up to 3.4 - 3.6 ± 0.9 mm per year

- see [L. 86]), respectively of the "Adriatic wedge" protruding to the north and shortening in the east-west direction. Several historical earthquakes occurred in the area with M > 6 macroseismic magnitude, the strongest of which was recorded on the Idrija Fault in 1511 (estimated magnitude M = 6.8).

The occurrence of earthquakes in central Europe (incl. the Temelín NPP region) with magnitudes of M ≥ 5.3 (I_0 ≥ 8° MSK-64) in correlation with the course of seismogenic lines is shown in Fig. 20. Earthquakes are contemplated in the period of observation from the year 1000[30] to 2007; the data on focal locations and earth shock intensity were taken from [L. 132]. The letter "R" designates events that were reviewed during the study [L. 163]. The position of seismogenic lines, including the rating of the potentially generated magnitude, were taken from the study [L. 160].

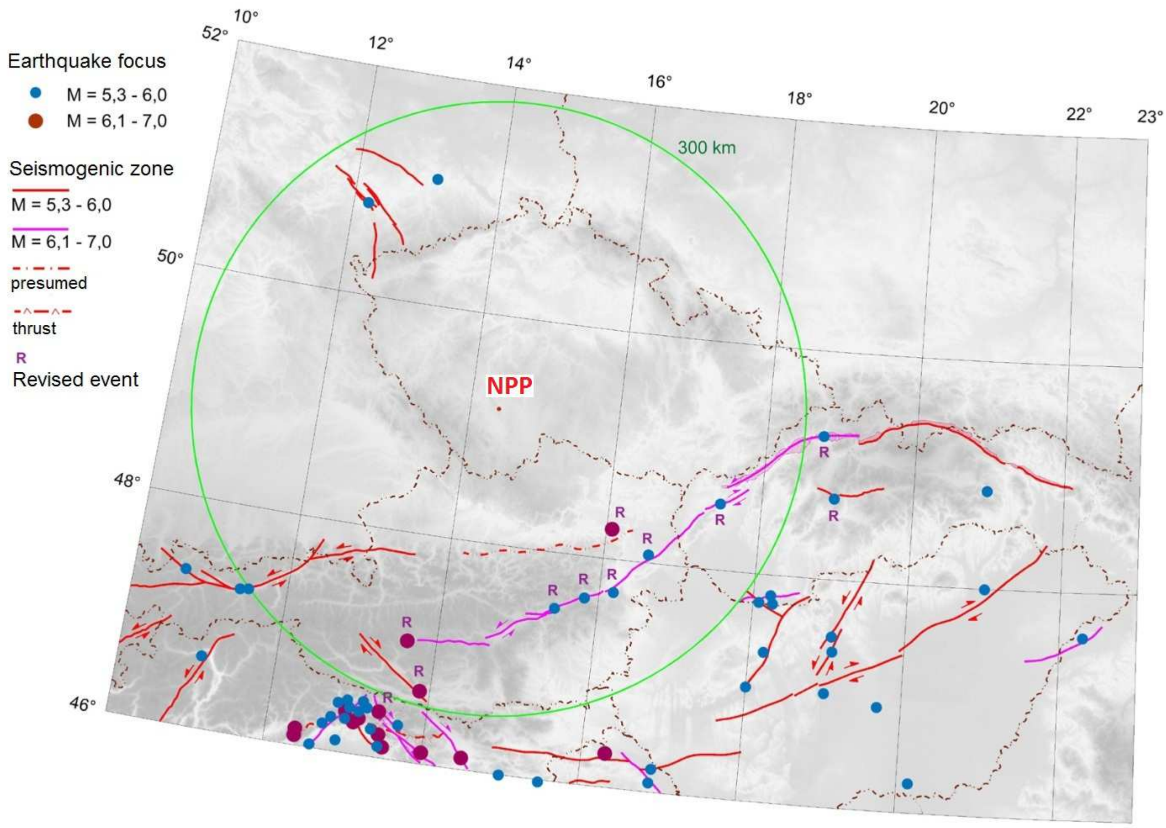


Fig. 20 Distribution of the foci of historical earthquakes with Mw ≥ 5,3 (I_0 > 8° MSK-64) in correlation with the course of seismogenic lines.

In the Bohemian Massif, increased seismic activity may be observed namely in the area of Kraslice – Aš – Vogtland in the western part of the region and in the area of Hronov – Poříčí in the northern part of the region. Less intense tremors were recorded in the southern part of the Moldanubicum, in particular, in the area of Kunějov or in the areas of Litschau and Pregarten near Linz in Austria. Catalogue data on close historical earthquakes and instrumentally recorded events occurring within a radius of approx. 70 km from Temelín NPP were closely reviewed in 2008

30 First written records in Bohemia.

(see basic material [L. 161]). The reviewed positions of earthquake epicenters and instrumentally recorded events are shown in Fig. 29.

The seismic load distribution in the territory of the Temelín NPP region may be best demonstrated on a map of ground motion acceleration (see Fig. 21) developed as a part of the GSHAP project [L. 87], - SESAME Peak Ground Acceleration Map [L. 115]).

In order to assess the seismic hazard for the Temelín NPP site, one should be familiar with the model of the geological environment, through which seismic waves spread, in addition to possessing knowledge of the distribution of earthquake foci and of the course of the main seismogenic lines.

The construction of the model, which intersects the Bohemian Massif in the NW-SE direction and then continues to the Carpathian system, was made possible due to the results of crust modelling obtained through extensive profile measurements carried out during the *Celebration 2000* seismic experiment (see e.g. [L. 174], [L. 101], [L. 102]).

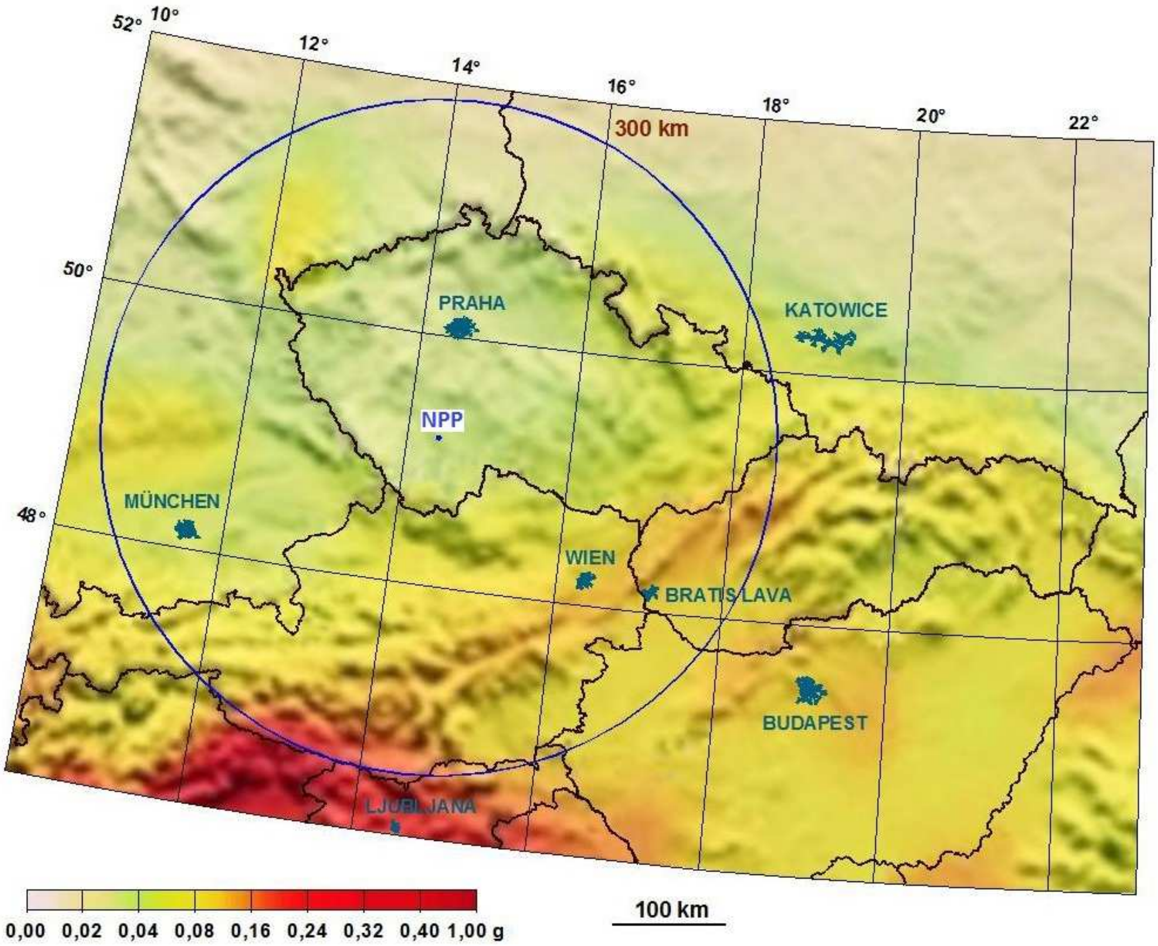


Fig. 21 Map of PGAH distribution for the reference return period of 475 years with 90% probability of non-exceedence within 50 years. Taken and modified from [L. 115].

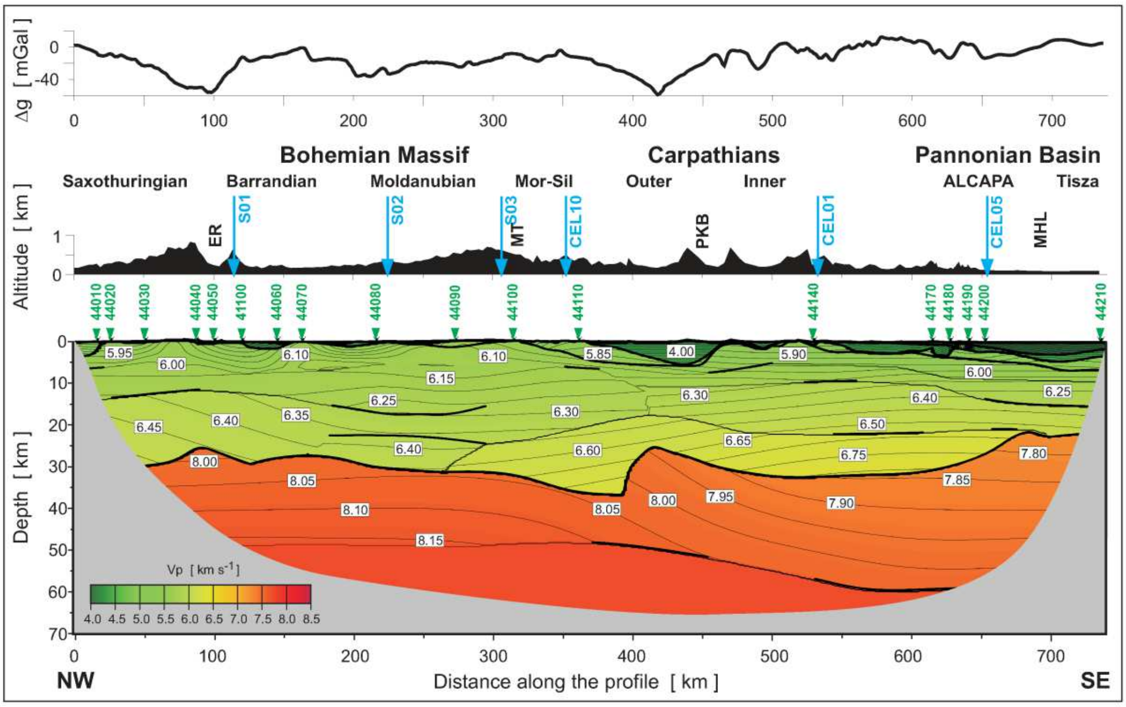


Fig. 22 Interpretation of seismic refraction data from the SUDETES 2003 project pertaining to the Bohemian Massif and its contact with surrounding units: P wave velocity model for the crust and upper mantle along the S04 profile [L. 102]).

Based on the S04 profile [L. 102]), a 1-D model of the environment was created (see Fig. 22) and subsequently used during a neodeterministic assessment of seismic hazard carried out with the aid of synthetic seismograms (see Section 2.6.7.1.6).

2.6.2.2 STABILITY OF THE GEOLOGICAL STRUCTURE OF THE NEAR REGION

2.6.2.2.1 Geological Development of the Near Region

The existing ETE1,2 and the construction site of ETE3,4 are located in the southern part of the Bohemian Massif, in an area belonging to the Moldanubian complex. The Moldanubian complex forms the crystalline fundament of the area and it is represented by both lithofacial units, i.e. of the Monotonous and Variegated Groups. The structure of the Moldanubian crystalline complex had been formed during multiple phases of ductile and fragile deformation until the end of the Paleozoic, whereas older structures were subject to repeated activation and transformation [L. 63]. The most widespread rocks include biotite, biotite-sillimanite or biotite-cordierite paragneisses and migmatites with occasional inserts of quartzites, amphibolites, granulites, and orthogneisses. These metamorphites are the product of complex polyphase, nappe-character deformation occurring in both the Cadomian and the Hercynian metamorphic and deformation cycles [L. 68]. The Hercynian deep reactivation of the older substrate also brought about intrusion of granitoid massifs, accompanied by extensive migmatitisation. In the north of the region of southern Bohemia, numerous offspurs of the Central Bohemian Pluton, represented by melanocratic amphibole-biotite syenites in the surroundings of Písek, Protivín and Vodňany, penetrate through the mantle of Moldanubian metamorphites. The central Moldanubian pluton, i.e. the Ševětín granodiorite, processes from the basement and along the western edge of the Třeboň Basin.

The subsequent tectonic development of the region of southern Bohemia was affected by two significant fault systems of the NNE-SSW direction (i.e. the direction of the Blanice Furrow) and the NW-SE direction (i.e the direction of the Hluboká Fault). Both fault systems had been formed by or before the final phases of the metamorphosis of the Moldanubicum and they substantially influenced the formation and development of the platform cover of the area (see Tab. 108). In the pre-Carboniferous period, the area was intensely mobile and the tectonic activity, manifesting itself through inverse uplift movement, aided in the following extensive denudation of the paragneiss mantle of the Ševětín granodiorite. The release of tangential stresses associated with the Asturian tectogenic phase resulted in the tectogenic predisposition of the depression structures in the crystalline basement. The depressions formed during synsedimentary subsidence were filled with continental sediments particularly in the Stephanium C and Lower Autunium period. The activity of the younger transverse fault system caused the formation of transverse elevations and depressions, as well as occasional sedimentation interruptions in the elongated structure of the Blanice Furrow. In consequence of the tectonic activity of transverse faults and the subsequent Mesozoic denudation, Permo-Carboniferous sediments in the Blanice Furrow appear as separated, tectonically bounded blocks. The compression of the Permo-Carboniferous blocks in the youngest phases of the Hercynian orogeny induced the formation of overthrusts and irregularities in the course of anthracite seams (see report in the publication [L. 70]). The youngest tectonic movements in the Permian block occurred during Saxonian tectogenesis.

After the Hercynian phases had subsided towards the end of the Permian, the south Bohemian area of the Moldanubicum became a part of the consolidated epi-Hercynian platform. In the early Mesozoic, orogenic processes were replaced by deep denudation and planation of the Hercynian mountain chain. During the Upper Cretaceous, the area was a part of a low peneplain lying at a very low elevation above the level of the Paratethys Sea.

In connection with the commencement of the Mediterranean tectogenic phase in the neighbouring Alpine-Carpathian area, the tangential stresses affecting the southern part of the Bohemian Massif from the SW and S are released and initiate the revival of tectonic activity [L. 168]. Concurrently with the tectogenic development of the Upper Cretaceous sedimentation area in the Eastern Alps and their foreland, an area of sedimentation is formed in southern Bohemia. Faults aligning in the NW-SE direction largely contribute to its formation, namely in the area between Vodňany and České Velenice and onwards to Waldviertel, Austria. Depressions are created in the block-segmented original peneplain where fluvial-lacustrine and wash sediments deposit. Their source material is a crystalline complex exposed to deep kaolinic weathering [L. 80]. Analogically to the Lower Gosau layers in the Alps [L. 128], the onset of the sedimentation phase is dated to the Coniacian or the Lower Santonian.

Two centres were created in the sedimentation area extending in the NW-SE direction - i.e. the České Budějovice Basin and southern part of the Třeboň Basin, which was drained towards the south-east. However, a comparison of the individual sedimentation cycles in different parts of the basins indicates that the syngenetic subsiding movement was identical in the basins and therefore, both have gone through similar tectonic development [L. 181].

Sedimentation of the Lower unit of the Klikov Formation (K_2^1) was interrupted in the southern part of the Třeboň Basin by more marked tectonic movements on the Nové Hrady Fault and the Stropnice line. The change in the gradient conditions resulted in the formation of the so-called "margin facies" between Třebeč and Nové Hrady, composed of coarse-grained sediments with pebbles of quartz and granitoids with size ranging from 5 to 7 cm. The main types of sediments of the Klikov Formation include coarse kaolinic sandstones, fine-grained conglomerates, dark grey sandstones and claystones with plant debris, or variegated claystones. The thickness of the Lower unit of the Klikov Formation ranged from 320 to 340 m.

The Ilsede phase in the Alpine area at the boundary of the Lower and Middle Santonian reflected itself in a uplift of the region of Southern Bohemia, in an interruption in the sedimentation of the Lower unit of the Klikov Formation and in a new configuration of the sedimentation area, particularly embracing the Třeboň Basin where sedimentation of the Upper unit of the Klikov Formation occurred (K_2^2). Another uplift of the southern part of the Bohemian Massif during the Wernigerrode phase of the Alpine folding in the late Santonian ended Senonian sedimentation in the basins of southern Bohemia. The moderate and uneven rise of the Bohemian Massif combined with the kaolinic weathering of crystalline rocks, as well as denudation, and planation had probably continued until the end of the Eocene. It is presumed that the Pyrenean phase later influenced the partial segmentation of the surface and the local sedimentation of the Lipnice Formation, the estimated age of which is dated to the Lower Oligocene (PG_3).

The moderate rising of the southern part of the Bohemian Massif accompanied by denudation and planation gradually subsided, and the Savian phase in the Lower Miocene (Ottnangian to Karpatian) induced the formation of a tectonically insignificantly bounded depressed area in southern Bohemia, which exceeded the borders of the Senonian basins. Less significant sediments of the Zliv Formation (N_1^1) consisting of only several-metre-thick layers of sandy clays and unsorted sandstones or fine-grained conglomerates deposited in the depressed areas. Towards the end of the Ottnangian, a pronounced uplift of the southern part of the Bohemian Massif ended the sedimentation of the Zliv Formation and caused its subsequent extensive denudation.

By the end of the Lower Miocene, the relief was once again rejuvenated due to the influence of the old Styrian phase. The subsidence of the wider area led to the formation of a vast depression extending far beyond the border of the south Bohemian Senonian basins. Through-flow lakes were created in the depressions, which were oriented in the NW-SE and NNE-SSW directions and drained towards the south. Fluvial-lacustrine sedimentation commenced in the Karpatian with the deposition of the lower Mydlovary Formation (N_1^{2a}); the Pištín "ditch" in the České Budějovice Basin and the Soběslav and Šalmanovice "ditches" (Stropnice line) in the Třeboň Basin, and gradually extended to the wider periphery of the Senonian basins. The lithological development of the sediments of the Mydlovary Formation was primarily influenced by the oscillation of the elevation of the sedimentation area relative to the level of the nearby sea. The onset of sedimentation was accompanied by the deposition of basal, sandy gravel and grey-green clayey sands that changed to greenish clays in the top strata. During the periods of shallowing, autochthonous lignite seams accompanied by freshwater clay-diatomite sediments developed in the basins and along their edges.

The complete release of pressures in the southern part of the Bohemian Massif associated with the tectogenesis of the young Styrian phase led to the subsidence of the area to the level of the Paratethys Sea and to the consequent penetration of brackish and sea water into the area of sedimentation [L. 175]. Rising water levels in the sedimentation area ended the development of autochthonous seams and caused a change of the lithological character of the sediments. Frequent sediments include grey-green, brown or beige diatomite clays with montmorillonite, which belong to the Upper unit of the Mydlovary Formation (N_1^{2b}).

The young Styrian phase, which had reflected itself in a slight uplift of the southern edge of the Bohemian Massif, brought about the end of the sedimentation stage of the Mydlovary Formation and its subsequent partial denudation. In the Badenian, a shallow sedimentation area was formed namely in the southern part of the Třeboň Basin where lacustrine freshwater clays with a high share of montmorillonite, as well as diatomite clays of the Domanín Formation (N_1^3) were deposited.

Their contemporary equivalents in the České Budějovice Basin and its periphery are the Vrábče beds with "in situ" moldavites (N_1^{3V}) and the Koroseky gravel sands (N_1^{3K}) with slightly abraded moldavites [L. 198].

As a result of another uplift of the basin area in the late Badenian, Miocene sedimentation in Southern Bohemia completely disappeared. Presently, the whole area was exposed to kaolinic weathering, partial denudation of older deposits, and overall planation.

Following the Rhodanian phase of the Alpine folding in the Middle Pliocene, the original Miocene basins and their peripheries experienced fluvial-lacustrine sedimentation of the Ledenice Formation (N_2) as a result of the subsidence of the South Bohemian region. The sedimentation of the Ledenice Formation ended with the onset of the younger Pliocene tectogenesis that reached its climax during the post-Dacian phase. It resulted in the domal and tectonic uplift of the southern parts of the Bohemian Massif (Blanský Forest [Blanský les], Nové Hrady Mountains [Novohradské hory] and Nová Bystřice Uplands [Novobystřická pahorkatina]) and in a change in the drainage pattern of the South Bohemian region to its present-day northbound direction, i.e. to the foundation of the existing Vltava valley. Moreover, the activity manifested itself in more intense denudation of the older sediments throughout Southern Bohemia. Deposits of the Ledenice Formation and Miocene sediments were considerably reduced or even completely denuded at some locations across the basin.

The final phase of Tertiary sedimentation, which is dated to the Romanian phase, brought about the deposition of predominantly fluvial, sand and gravel sand sediments (see lit. [L. 198]) - Kamenný Újezd gravel sand ($N_2^{KÚ}$). The deposition of these sediments probably occurred in the river system, which was already drained to the north.

A dominant role in the final shaping of the morphology of the southeastern part of the České Budějovice Basin was played by the Paleo-Vltava, the Paleo-Blanice in the northwestern part, and by the Paleo-Lužnice in the Třeboň Basin.

Tab. 108 Geochronological table (assembled with the aid of [L. 120]).

Geochronological scale			Age (Ma)	Type of sediments (stratigraphic unit)	Thickness (m)
HOLOCENE - Q			0,01	Alluvial deposits	3
PLEISTOCENE	Würm - Q		0,12	Terrace sediments sands - gravels	6
PLEISTOCENE	Riss - Q		0,30		
PLEISTOCENE	Mindel - Q		0,78		
PLEISTOCENE	Günz - Q		1,60		
PLEISTOCENE / PLIOCENE	Donau - Q		1,85		
PLIOCENE	Romanian		3,7	Kamenný Újezd gravels - $N_{2\,KÚ}$	5 - 7
PLIOCENE	Dacian		5,8	Ledenice Formation - N_2	25
MIOCENE	Pontian		8,8		
MIOCENE	Pannonian		11,5		
MIOCENE	Sarmatian		13,7		
MIOCENE	Badenian	Upper	16,8	Koroseky sandy gravels - N_{1K}^{3} Vrábče sands to clays - N_{1V}^{3}	20
				Domanín Formation - N_1^3	35
MIOCENE	Badenian	Middle		Mydlovary Form. Upper unit - N_1^{2b}	85
MIOCENE	Badenian	Lower		Mydlovary Form. Lover unit - N_1^{2a}	
MIOCENE	Karpatian		17,6	Zliv Formation - N_1^1	25
MIOCENE	Ottnangian		19,0		
MIOCENE	Eggenburgian		22,2		
MIOCENE	Egerian		25,0		
OLIGO-CENE	Rupelian		≈ 35		
OLIGO-CENE	Lattorfian		≈ 38	Lipnice Formation - PG_3	40
EOCENE			≈ 55		
PALEOCENE			≈ 65		
U. CRETACEOUS – Senonian	Maastrichtian		≈ 73		
U. CRETACEOUS – Senonian	Campanian		≈ 83		
U. CRETACEOUS – Senonian	Santonian	Upper	≈ 88	Klikov Formation Upper unit - K_2^2	140
U. CRETACEOUS – Senonian	Santonian	Middle			
U. CRETACEOUS – Senonian	Santonian	Lower		Klikov Formation Lower unit - K_2^1	320
U. CRETACEOUS – Senonian	Coniacian		≈ 93		
U. CRETACEOUS	Turonian				
PERMIAN	Autunian	Upper	270 -	Chýnov beds - Pa_1^{1-2}	80
PERMIAN	Autunian	Lower	286	Lhotice beds - Pa_1^1	160
CARBONI-FEROUS	Stephanian	C	286	Peklov beds - PaCs	100
CARBONI-FEROUS	Stephanian	B	-	Moldanubicum	
CARBONI-FEROUS	Stephanian	A	296		
CARBONI-FEROUS	Westphalian (A - D)				
CARBONI-FEROUS	Namurian (A - C)				

As regards the accumulated proluvial gravelly sands or sandy gravels ($^{p}{}_{šp}Q$) from the surroundings of Zbudov, i.e. the so-called Zbudov gravel sand, their dating remains rather unclear (Upper Pliocene–Riss).

The uplift tendencies had prevailed from the Upper Pliocene until gradually disappearing in the older Pleistocene. They resulted in the further denudation of the sedimentary filling of the České Budějovice Basin and in the deepening of the valley network in the Písek Uplands [Písecká pahorkatina], which projected from the basin as a marked elevation.

The gradual deepening of the Vltava valley in the Upper Pliocene and the Pleistocene induced the formation of river terraces (2 Pliocene, 6-7 Pleistocene). Loess banks ($^{e}Q_{W}$) were created locally during the Würmian period. The features documenting the updoming of some relief segments, as well as the possible decaying tectonic uplift of the "border mountains" are represented by the moderate inclination of the Günz and Mindel terraces in the České Budějovice Basin in westward cross profile and the sloping of the marshes in the Třeboň Basin towards the east [L. 82].

2.6.2.2.2 Potential of Tectonic Movements in the Near Region

The current morphology of the region of Southern Bohemia where the near region of Temelín NPP is situated is the result of long-term geological development, to which tectonic, depositional and erosion phenomena have contributed. A fundamental role in the development of the South Bohemian region is attributed to the Alpine folding, the individual phases of which affected the face of the local relief and reflected themselves in the tectonic activity of the Hercynian and older fault systems along the edge of the Bohemian Massif.

The most pronounced manifestation of the Saxonian tectonic activity of faults in the South Bohemian region is the constitution of two basin structures - i.e. the České Budějovice and Třeboň basins on the intersection of the fault systems aligned in the NNE-SSW and NW-SE direction.

The last intensive movements on the faults that were associated with the upraise of the southern edge of the Bohemian Massif (outside the near region of Temelín NPP) appeared in connection with the post-Dacian phase of the Alpine folding. Tectonic activity was the most intense in the south (in the border mountains - e.g. on the Nové Hrady Fault).

The uplift of the southern part of the Bohemian Massif had continued at least to the early Pleistocene. In this period, erosion prevailed over accumulation and the basin sediments were intensely removed, while the crystalline basement formed by Miocene sediments was largely exhumed and the relief was subject to expressive morphological diversification. The Miocene sediments were only preserved in the deepest depressions and in the bottoms of pre-Miocene fluvial river beds.

Neotectonic movements (i.e. updoming) had probably been still active at the boundary of the Pliocene and the Pleistocene, and they have been dying away during the Pleistocene. It is presumed, however, that their impact on relief changes and fault movement was only minor.

It is beyond any doubt that the long geological development of the region of southern Bohemia saw the formation of a number of fault structures, which may be traced in the geological structure of the region. An overview of major tectonic lines in the near region of Temelín NPP, which are plotted in official 1 : 25,000 (or 1 : 50,000) geological maps and discussed in the study [L. 187]), is shown in Tab. 109.

The Hluboká, Drahotěšice, Rudolfov, or the Munice Fault, as well as the presumed Hrdějovice Fault are associated with the bounding or disruption of older sedimentary units of the platform cover, e.g. the Permo-Carboniferous relics of the Blanice Furrow and the Senonian basin filling. While the bounding of Miocene sedimentation in the Pištín Ditch is attributed to the faults with the assumed faults of Zbudov and Haklovy Dvory. Nevertheless, it has not been demonstrably proven whether these two faults disrupt the crystalline fundament in the basement of the České Budějovice Basin or not[31]. According to our presumptions, the remaining faults represent fault structures that are linked to the local crystalline fundament (old fault structures) and that could have predisposed the formation of river beds of the Tertiary fluvial system without interrupting their later filling.

Tab. 109 Overview of faults identified in the near region of Temelín NPP according to [L. 187].

Fault	Directional course of fault	Minimum distance from ETE3,4
Vodňany Mylonite Zone	NE-SW	4 km
Líšnice Fault	NNE-SSW	7 km
Hluboká Fault	NW-SE	9 km
Radomilice Fault (assumed)	N-S	10 km
Vlhavy Fault (assumed)	N-S	10 km
Munice Fault	NNE-SSW	10 km
Zbudov Fault (assumed)	NW-SE	11 km
Blanice Valley Fault	N-S	12 km
Drahotěšice Fault	NNE-SSW	13 km
Haklovy Dvory Fault (assumed)	NW-SE	15 km
Hrdějovice Fault (assumed)	NNE-SSW	19 km

With a view to the apparent morphological manifestation and other features supporting the suspicion of recent capability of the fault, particular attention was given to faults in the so-called Pořežany Ditch, the Vodňany Mylonite Zone, and to the N-S faults along the western edge of the near region – the Vlhavy Fault, the Blanice Valley Fault and especially to the Hluboká Fault in the NW-SE direction during several phases of the evaluation of fault structures in the near region of Temelín NPP.

The faults of the Pořežany Ditch (see basic material [L. 187]) were given priority attention namely due to their proximity to Temelín NPP and to the bounding of Pliocene sediments by faults with horizontal shifts reaching dozens of metres (see geological map [L. 180]). In connection therewith, the study [L. 187] focuses in greater detail on the question of the disruption (bounding by faults) of similar relics of Miocene sediments caused by capable faults. Nonetheless, investigations (including geophysical measurements, drilling and trenching) identified no tectonic bounding ordisruptions of the relics in the Pořežany Ditch that could be attributable to faults. By applying these conclusions also to other Miocene sediment relics in the near region of Temelín NPP, it was possible to confirm the hypothesis presented in [L. 69] and [L.

[31] Instead, the results of seismic reflection profiling carried out by the AIP project team (Austrian Interfacing Project) - see lit. [L. 78] on the Zbudov Fault near Mydlovary seem to indicate the absence of any disruption of the basement. At the same time, the profile does not show any clear reflectors that would identify the line of the Zbudov Fault.

189], which claims them to be denudation remnants of the filling of old Tertiary river valleys.

Furthermore, research was carried out on the morphological slope intersected by the so-called Vodňany Mylonite Zone defined in [L. 127] and [L. 194]. For this purpose, a slope near the municipality of Fanfíry situated approximately 6 km from Temelín NPP was selected. The exploratory trench at the base of the slope uncovered cataclased migmatites and Moldanubian paragneisses. Nevertheless, no apparent fault structure separating two different facies of crystalline rock or tectonic contact of the crystalline complex with Miocene sediments was revealed at the base of the slope.

An evaluation of the Blanice Valley Fault and of the Vlhavy Fault was effected in two phases in the years 2010 and 2012 (see basic materials [L. 164] and [L. 167]). The fault at the western edge of the near region of Temelín NPP is geographically associated with Miocene sediments of the Mydlovary Formation. As in the case of the Pořežany Ditch, the aim was to verify the presumption that older fault tectonics or intense fracture zones in the area had created favourable conditions for the formation of depressions subsequently filled with Tertiary sediments (see basic materials [L. 189], [L. 177] and [L. 83]). In this area, it is namely the tectonically weakened zone aligned in the N-E direction that runs from Čičenice via Protivín and Tálín Pond to Nový Dvůr.

The movement activity on the Hluboká Fault was verified with the aid of paleoseismic methods in the years 2009 and 2010 within the framework of the Science and Research Project commissioned by the SÚJB based on the results of discussions of the Czech-Austrian parliamentary commission concerning the safety of Temelín NPP [L. 188].

The Hluboká Fault was studied in-depth in the most exposed section between Munice and Vráto, i.e. along a 13-kilometre long segment characterised by the most pronounced geological, geophysical, and morphological manifestations. Near the municipality of Munice, the NW course of the Hluboká Fault is interrupted by the Munice Fault, which is aligned in the N-S direction and indicated in the basement of the České Budějovice Basin by an almost 200 m high throw of the Senonian filling base.

The present morphological manifestation of the Hluboká Fault, which was considered to be the result of young vertical movement on the fault by some geologists, has shown to be the product of differential erosion in the zone of contact of very lightly cemented sediments of the basin filling with stable rocks of the crystalline complex. The straight running and undoubtedly tectonically predisposed slope was exhumed during the Pleistocene by the rapidly deepening Vltava River. Due to river erosion, the slope receded and the trace of the fault defined by the contact of the rocks of the crystalline complex with the filling of the České Budějovice Basin runs several metres from the base of the slope inside the basin platform.

Paleoseismic methods, including the dating of sediments that cover the trace of the fault and are undisturbed by its motion, prove that the surveyed section of the Hluboká Fault has remained tectonic repose for the past 22,000 years, at minimum. The results of the study of the Vltava terrace sediments and their depositional conditions on blocks separated by the fault indicate the absence of movement on the Hluboká Fault also during the Würmian period (i.e. over the past 100,000 years). Considering the geomorphological features of the relief of the České Budějovice Basin and its northern foreland, as well as the uninterrupted phased deepening of the

Vltava river bed in the Rissian and Mindelian periods, it is possible to presume with high probability the absence of recent tectonic movements on the faults also in the Middle Pleistocene (approx. 800,000 years).

The possibility of obtaining direct evidence of the extinction of movement activity on the Hluboká Fault (as well as other significant faults in the near region of Temelín NPP) in the older Pleistocene and the Pliocene is considerably complicated by the absence of sediments of the same age in coincidence with traces of these faults and their occurrence in isolated relics, the position of which is rather difficult to mutually correlate. When evaluating the potential capability of the faults in the near region of Temelín NPP in these geological epochs, we may only refer to indirect evidence, such as correlations of the bases and fillings of Miocene river channels, or correlations of the positions of the relics of the Zliv Formation, which is naturally encumbered by uncertainty.

Nevertheless, based on the correlations of the position of sediments of various ages ranging from the Miocene up to the Pleistocene, as well as based on other findings pertaining to the geological structure and the overall character of the territory, it is possible to presume with high probability that tectonic activity in the Pliocene, which is so apparent along the southern edge of the Bohemian Massif, did not manifest itself in the near region of Temelín NPP. The phenomena associated with Pliocene tectonic activity that were recorded in the vicinity of the near region of Temelín NPP are the result of creep during an uneven updoming of the segments of the geological structure of the region of Southern Bohemia.

2.6.2.3 BRIEF RECAPITULATION OF THE FINDINGS ON THE GEOLOGY OF THE TEMELÍN NPP CONSTRUCTION SITE

The geological conditions on the main construction site of Temelín NPP, i.e. in the area of the planned units 3 and 4 with VVER 1000 reactors, were appraised during several phases of engineering geological surveys carried out in the form of orientation, detailed and supplementary surveys between the years 1980 and 1989. The basic material and data were sorted and reviewed in the course of preparation of the "Contract Documents" [L. 159] for the MGU supplier for Temelín NPP. Additional, complementary information was obtained through supplementary surveys effected in the years 2008 and 2010 (see basic materials [L. 192] and [L. 165]). Based on the above-mentioned basic material and data, the following subsurface "profile" for the Temelín NPP construction site is presented within the meaning of Paragraphs 3.3 to 3.5 of the IAEA Safety Guide No. NS-G-3.6 [L. 13].

2.6.2.3.1 Rock Massif Characteristics

From a structural and geological viewpoint, the rock massif on the ETE3,4 construction site may be interpreted as a integral, unevenly surface-weathered geological block, which is disrupted by several discontinuity systems of local importance.

A dominant structural and tectonic feature of the massif on the Temelín NPP construction site (and the "Týn nad Vltavou crystalline complex" of the Moldanubicum Monotonous group) is the well preserved planar structure - i.e. foliation of the NE-SW direction - dipping towards the NW. Typical of this structure is multiple alternation of the schistose beds of migmatitised paragneisses and migmatites, with numerous mostly concordant injections of granitoid rocks. The planar folding structure is disrupted by dislocations that are largely aligned in the north-south direction.

The failures appear as a system of longitudinal, occasionally of non-associated and uneven discontinuities, bound to the occurrence of rigid rock bodies of limited extent (especially granitoids) that were exposed to tectonic stress equalizing, as manifested mostly by semi-plastic deformations of the lithological positions in the surrounding "softer" gneiss complex. The rocks in these zones, often several metres wide, range from intensely to very intensely jointed, weathered to highly weathered, with frequent secondary alterations, however, without continual and thicker filling (e.g. fault-gouge, etc.). Inside the gneiss massif, partial displacements may be observed with cross striation, slickensides, or zones with cataclased rocks. The geological age of these syntectonic failures is dated before the Upper Cretaceous.

These disturbances, however, do not represent regional geological lineaments that would disrupt the homogeneity of the Moldanubian block of the main construction site.

2.6.2.3.2 Lithological Characteristics and Stratigraphic Data

The bedrock of the Temelín NPP main construction site[32] is formed by Moldanubian metamorphites of the Monotonous group. The main rock types include sillimanite-biotite paragneisses and their equivalents migmatitised to various intensities. Their representation on the construction site amounts to 94%. Vein Variscan granitoids (granites - pegmatites), usually of small thickness are found at occasional locations on the construction site. They form approx. 6% of the basement rock mass (see basic material [L. 144]).

The rocks found on the construction site are designated by specific letter codes: A (Quaternary), B (Tertiary), C (Variscan magmatites), D (pre-Cambrian metamorphites) [L. 144].

In the Quaternary, 5 basic types of soils are differentiated: A1 (Topsoil), A2 (sandy silt), A3 (silty sand), A4 (sandy clay), A5 (silty-sandy gravel), A6 (hill debris), and A9 (man-made fill).

Tertiary sediments (B) were not found on the main construction site.

Hard rocks are classified according to rock types as follows: C1 (vein granite), C2 (pegmatitic granite), C3 (pegmatite), C4 (vein quartz), D1 (sillimanite-biotite (biotite) paragneiss), D2 (migmatized sillimanite-biotite (biotite) paragneiss), D3 (biotite greywacke gneiss), D4 (quartzite gneiss), D6 (biotite (sillimanite-biotite) migmatite). Zero is used to designate eluviums of C or D type rocks and 9 is assigned to tectonic zones.

The above described system of rock and soil classification was used during the compilation of the exploratory site database [L. 159], as well as in the documentation pertaining to the cores of new exploratory boreholes.

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[32] The main construction site is the area investigated within engineering geological surveys conducted in relation to the siting and construction of Temelín NPP consisting of 4 VVER 1000 units. At present, it embraces the whole guarded area of the current Temelín NPP, including the western foreland (outside the guarded area), which is designated for the placement of the cooling towers of the new nuclear installation at Temelín NPP.

2.6.2.3.3 Anthropogenic Interventions in the Land

The top layer of the territory designated for the placement of the crucial structures of ETE3,4 is formed by man-made fills comprising soils and loosened rock excavated from the foundation pits of NPP1,2. The fills covered the footing bottoms of the planned units 3 and 4. The executed interventions in the land, including the localization and documentation of covered structures built in connection with the preparation of the construction of units 3 and 4 with VVER 1000 reactors, were initially investigated with the aid of geophysical methods (e.g. man-made fill depth range - see basic material [L. 192]), and in 2010, in greater detail with the aid of geophysical methods combined with exploratory drilling - see basic material [L. 165]. The man-made fill cover a layer of mostly moderately weathered and slightly weathered rocks of the bedrock. Fills were also used for the levelling of the ground along the circumference of the construction site.

Basic materials and data, as well as geophysical measurements and the results of exploratory drilling in the northern part of S1[33] area indicate that the depth of earthwork has not exceeded 501 metres above sea level in the major part of the area under review. Several locations were identified where the earthwork reached greater depths. These mainly include excavations for cooling water pipe channels, collection tanks and drainage systems. We presume that these interventions in the subsoil may have reached approximately 498 metres above sea level (see basic material [L. 165]).

More significant interventions in the subsoil are represented by the excavation pits for the cooling water pumping stations in the southern part of S1 area. Here, anthropogenic interventions reach 495 metres above sea level and are rather extensive in spatial terms.

The material detected in the backfills mostly consists of excessive soils from rough ground shaping and the excavation pits of other structures. Groundwaters bound to the backfills are non-aggressive or slightly aggressive (aggressiveness stemming from water acidity and aggressive CO_2 content) - see report [L. 165].

2.6.2.3.4 Determination and Description of the Main Geotypes (Foundation Soil Profile)

Based on the results of older and newer investigations, the following geotypes may be determined on the ETE3,4 construction site:

- Quaternary cover
- Man-made fills
- Eluviums
- Completely to highly weathered rock (i.e. zone of prevalent rock weathering)
- Moderately or slightly weathered and fresh rock

A schematized "profile" of the foundation soil, with allowance for anthropogenic interventions in the subsoil, is shown in Fig. 23.

[33] Area designated for the placement of the main generating units of the new nuclear installation at Temelín NPP.

Quaternary Deposits

Quaternary **Deposits** are not characterised in greater detail as they were removed during rough ground shaping. A description of the characteristics of the removed rock and soil layers is provided in [L. 144]). The significance of the removed layers is not decisive for the evaluation of the ETE3,4 structure foundations.

It should be mentioned, however, that in the course of rough ground shaping on the main construction site of Temelín NPP, the top layer of the cover was excavated to a depth of 6 metres. The original Quaternary cover and the fossil weathering zone (eluvium) were removed. Further changes were induced by the digging of the footing bottoms for VVER 1000 units 3 and 4. The excavation pits were subsequently covered by backfills.

Man-made fills

The **Man-made fills** are formed by rock and soil material from the rough ground shaping of the Temelín NPP construction site.

The backfill material consists of eluviums of biotite and sillimanite-biotite paragneisses, as well as completely weathered to moderately weathered paragneisses. Based on a textural analysis and subsequent classification (see basic material [L. 165]), the materials embrace silty sands, clayey sands or sandy clays (sporadically), and sandy gravels.

The largest filled areas may be found in S1 area. Their thickness ranges from 0.5 m in the northwestern part of S1 area to almost 7 m in excavation pits for the cooling water pumping stations of VVER 1000 units 3 and 4 in the southern part of S1 area. In terms of elevation, we may presume that the base of the backfills in the northern part of S1 area lie at elevations of 501 - 502 metres above sea level. Deeper interventions in the subsoil are limited to the routes of the service water system channels (up to approx. 499 metres above sea level). In the southern part, the base of the backfills on the site of the former excavation pits lies at 495.15 metres, 497.16 metres and 491.20 metres above sea level (For details see basic material [L. 165]).

Eluviums

Eluviums were discovered in S1 and S2 areas only in several boreholes drilled during new explorations [L. 165]). The layer was largely removed from the area and it has been preserved only along the edges of S1 and S2 areas. Older basic materials and data (particularly [L. 144]) suggest that eluviums reach 495 - 496 metres above sea level along the southern edge of S1 area, while in its central and northern parts, they may have been preserved up to 500 metres above sea level. As regards S2 area, namely its northern and southern edge, eluviums may be found at elevations from 489 to 491 metres above sea level.

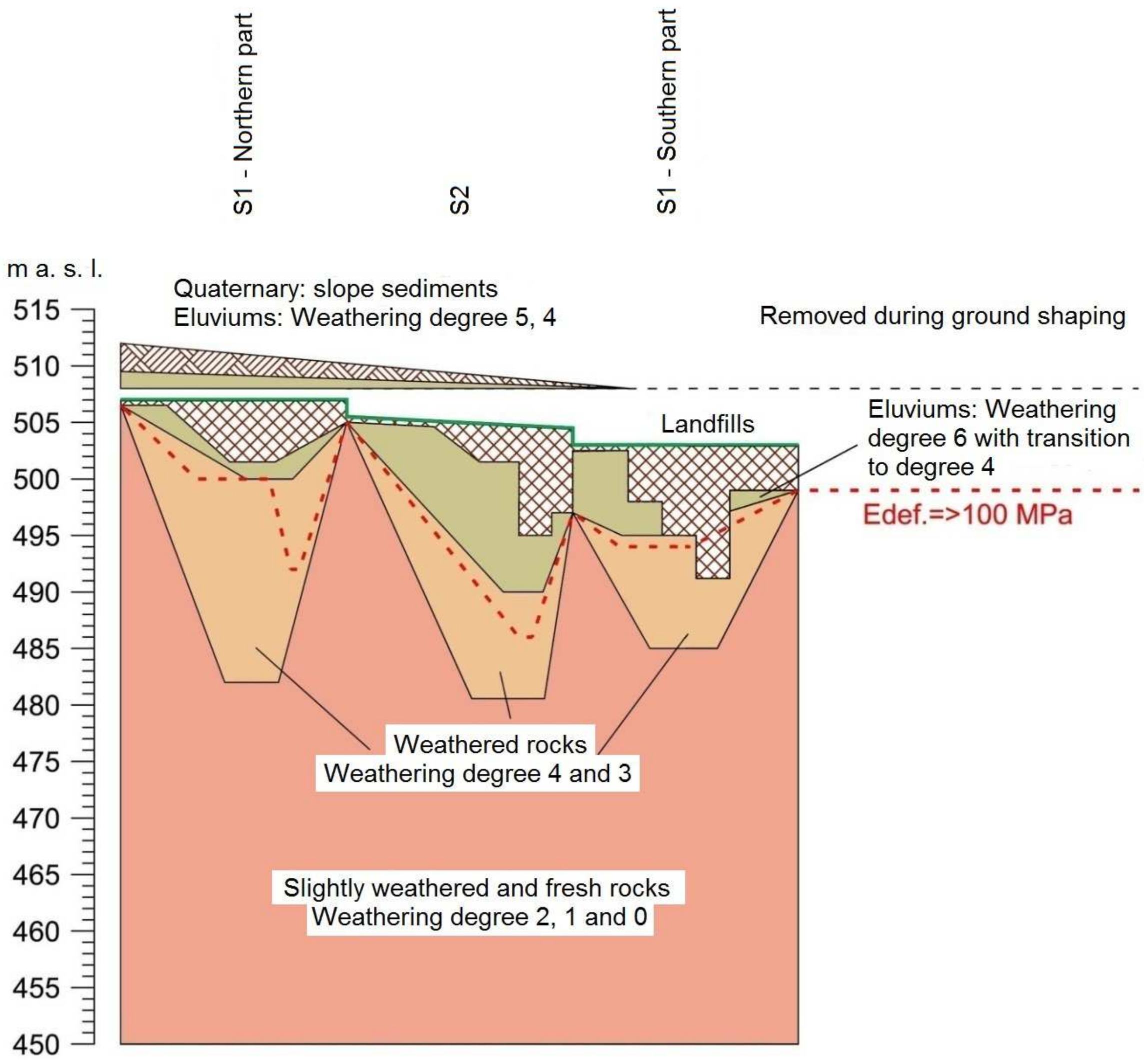

Fig. 23 Schematized "profile" of the foundation soil for S1 and S2 areas of the ETE3,4 construction site.

Completely to Highly Weathered Rock

The "zone of predominant rock weathering" usually ranged to depths from 5 to 15 m below the original surface of the territory. At depths starting from 20 metres below the surface, it changed into a zone of less weathered and fresh rocks.

The course of base contours of the eluvium and the "zone of predominant rock weathering" (see particularly basic material [L. 144]) evidence the uneven effect of weathering processes, depending on the tectonic disruption of the rock mass and the degree of the lithological resistance of rocks to weathering (i.e. silicification, solid quartz tongues, etc.).

The determination of the area of the maximum range of the zone in the subsoil of the central part of S2 area was the subject of a supplementary survey carried out in 2010 [L. 165]). Based on the survey results, mostly weathered rocks form the filling of small-sized depressions in the unweathered bedrock. In the central part of S2 area, two depressions were identified where weathered rocks reach up to 492 metres above sea level, respectively 488 metres above sea level. And they extend to approx. 480 metres above sea level along the edges of the area as indicated by [L. 144].

The same basic material suggests that weathered rock may occasionally appear at elevations from 482 to 485 metres above sea level.

Weathered rocks are generally classified as sandy soils (clayey sands), sporadically gravelly, as well as sandy clays or clayey and/or sandy gravels.

Moderately or Slightly Weathered and Fresh Rock

As mentioned above, the zone was found 20 metres and more below the original surface on the Temelín NPP construction site. At some locations, the zone is covered by a backfill layer (i.e. at an elevation of 502 metres above sea level or higher). And in depressions, it lies at the levels specified in the description of the zone of weathered rock.

2.6.2.3.5 Rock and Soil Properties

In the course of investigations, the individual rock and soil types were subjected to in-situ and laboratory tests in accordance with the then valid CSN technical standards. Tab. 110 below contains an overview of the physical and mechanical properties of the rocks and soils found on the main construction site. The indicated data were obtained through statistical evaluations of the results of laboratory rock and soil tests that were performed during geotechnical surveys carried out on the construction site in the 1980s. The data represent estimated values or value ranges of the main geotechnical parameters of the rocks and soils present on the construction site.

Tab. 110 Physical and mechanical properties of soils and rocks on the Temelín NPP main construction site (according to basic material [L. 145]).

Physical properties			Quaternary	Eluvium	Completely weathered rock	Weathered rock - fresh
Natural water content	W_n	[%]	20	18	-	-
Liquid limit (Atterberg)	W_{LA}	[%]	48	40	-	-
Liquid limit (Vasilyev)	W_{Lv}	[%]	44	37	-	-
Plastic limit	W_p	[%]	30	25	-	-
Plasticity index (Atterberg)	I_{pA}	[%]	18	15	-	-
Plasticity index (Vasilyev)	I_{pV}	[%]	14	12	-	-
Porosity	n	[%]	32.7	30.9	18.5	11.1
Void ratio	e	[-]	0.486	0.447	0.227	0.123
Bulk density (natural moisture)		[kg.m^{-3}]	2000	2240	2270	2450
Bulk density (dry)		[kg.m^{-3}]	1850	1900	2200	2400
Specific density		[kg.m^{-3}]	2750	2750	2700	2700
Effective angle of internal friction	φ_{ef}	[^{o}]	28	32	-	-
Effective cohesion	c_{ef}	[kPa]	20	17	-	-
Deformation modulus	E_{def}	[MPa]	10	20	30-35	100-300
Filtration coefficient	k	[m.s^{-1}]	1.10^{-7}	1.10^{-6}	-	-
Fatigue strength coefficient (sat.)	R_S	[MPa]	-	-	5	10
Softening coefficient	K_{RZ}	[-]	-	-	0.75	0.75
Weathering coefficient	K_{VS}	[-]	-	-	0.85	0.85
Consistency index	I_c	[-]	1	1.46	-	-

Newly conducted engineering geological surveys [L. 165], which included rock and soil laboratory testing and logging, confirmed the validity of the conclusions of the previous surveys of the Temelín NPP main construction site (see basic materials [L. 144] to [L. 150]). The key drill core was submitted to the archives administered by the Czech Geological Survey of Geofond Prague.

Based on the supplementary survey results [L. 165], it is also possible to state that all the geotechnical rock and soil parameters, the sequence and positioning of the individual lithological units and their layering, including the determination of the level of the layer surface of the specific deformation modulus, may be considered reliable.

2.6.2.3.6 Stability of the Foundation Soils under Static Load

When determining the stability of the foundation soil subjected to static load, we mainly start from the E_{def} value of the modulus of deformation. This value was determined during surveys of the main construction site (see reports [L. 144] to [L. 150]) by means of a statistical selection of the results of presiometric tests (or plate load tests) with respect to the specific rock type and degree of weathering. In general, the following values of the deformation modulus (see Tab. 111) may be assigned to the individual types of the present, variously weathered rocks.

The above specified "key" for the determination of the E_{def} deformation modulus value is deemed reliable and it is proposed as a guide for field evaluations of the stability of the foundation soil under static load.

Based on the previously conducted surveys and all supplementary surveys, it is possible to say that with respect to the area contemplated for the construction of the ETE3,4 Units, a ≥ 100 MPa deformation modulus may be considered from the depth of 495.0 metres above sea level. Zones (areas) with higher rock failure appear in the area contemplated for the location of the cooling towers. In zones with the deepest identified range of rock failure and weathering, the deformation modulus of E_{def} ≥ 100 MPa may be expected approximately from the depth of 494.1 metres above sea level.ETE3,4

Tab. 111 Deformation modulus values for the present rock types

Rock group	Rock type	Degree of weathering	E_{def} (MPa)
Quaternary	A2, A3, A4	-	8
	A5, A6	-	12
Pegmatitic granites	C1, C2, C3 (C0) C9	Decomposed Decomposed and completely weathered	20
	C1, C2, C3 C9	Completely weathered Highly weathered	35
	C1, C2, C3 C9	Highly weathered Moderately weathered	150
	C1, C2, C3 C9	Moderately weathered Slightly weathered to fresh	300
	C1, C2, C3, C4	Slightly weathered to fresh	450
Paragneisses to migmatites	D1, D2, D3, D6, (D0), D9	Decomposed Decomposed and completely weathered	15
	D1, D2, D3, D6 D9	Completely weathered Highly weathered	30
	D1, D2, D3, D6 D9	Highly weathered Moderately weathered	120

Rock group	Rock type	Degree of weathering	E_{def} (MPa)
	D1, D2, D3, D6 D9	Moderately weathered Slightly weathered to fresh	250
	D1, D2, D3, D4, D6	Slightly weathered to fresh	400

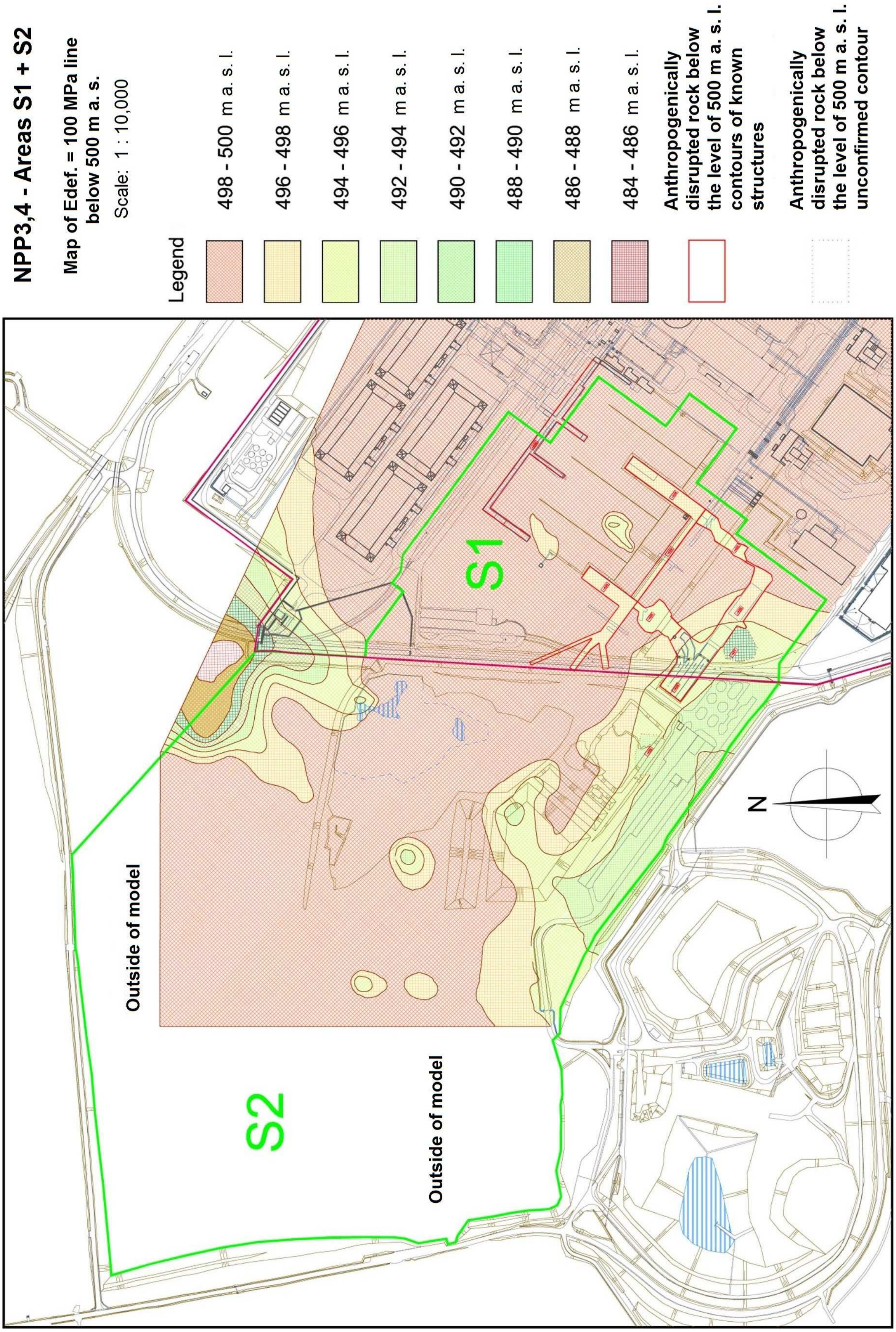

Fig. 24 Map of the distribution of E_{def} ≥ 100 MPa in plots S1 and S2 below the elevation of 500 metres above sea level.

2.6.2.3.7 Stability of the Foundation Soils under Dynamic Load

When determining the stability of the foundation soils subjected to dynamic load, we start from the results of logging (acoustic wave logging), which was executed in four boreholes as a part of the supplementary engineering geological survey conducted in 2010 (see basic material [L. 165]). The basic geomechanical parameters for the individual types of the present rocks were derived from the logging results. Tab. 112 below contains (minimum, maximum, median) values for the selected parameters (DENA - specific density; SIGS_K – unconfined compressive strength; POIS_A – Poisson's ratio; VS_ALT – shear wave velocity; ED_ALT – Young's modulus; GD_ALT – shear modulus) depending on the degree of metamorphic rock weathering. The parameters of other rock types present on the construction site are not specified due to their limited set of representative values.

Tab. 112 Basic geomechanical rock parameters (according to basic material [L. 165]).

Parameter		Variable	DENA	SIGS	POIS_A	VS_ALT	ED_ALT	GD_ALT
Rock	DW		(g/cm^3)	(MPa)	(-)	(m/s)	(GPa)	(GPa)
BiPg	0	Median	2.60	40.83	0.233	1976	25.00	10.14
		Min	2.44	39.26	0.206	1961	24.69	9.97
		Max	2.72	46.12	0.284	2038	26.98	10.51
BiPg	1	Median	2.54	45.66	0.258	1923	24.43	9.72
		Min	2.27	20.64	0.171	1834	22.26	8.98
		Max	2.73	59.78	0.339	2121	27.51	11.01
BiPg	2	Median	2.44	31.69	0.292	1913	22.68	8.94
		Min	2.27	13.75	0.175	1779	21.25	8.30
		Max	2.65	61.46	0.355	1968	24.50	9.52
BiPg	3	Median	2.29	37.5	0.222	1892	20.68	8.45
		Min	2.17	14.98	0.171	1798	19.23	7.61
		Max	2.68	56.01	0.30	2121	27.51	11.01
BiPg	4	Median	2.17	29.24	0.233	1817	17.18	7.25
		Min	1.86	15.39	0.132	1407	11.27	4.63
		Max	2.72	50.86	0.337	1980	21.72	9.03
BiPg	5	Median	2.15	24.06	0.258	1485	11.20	4.54
		Min	2.06	15.19	0.187	1388	10.82	4.20
		Max	2.31	25.18	0.281	1502	12.86	5.15

BiPg - biotite paragneisses and migmatized biotite paragneisses; DW – degree of weathering according to CSN-EN-ISO-14689-1.

The velocity of shear (S) waves, see variable VS_ALT in the table above, was used for the site categorization in compliance with the recommendations specified in Paragraph 3.1 of the IAEA Safety Guide No. NS-G-3.6 [L. 13] (site categorization for the purpose of seismic response analysis). According to the IAEA Safety Guide No. NS-G-3.6 [L. 13] (site categorization for the purpose of seismic response analysis). According to the AIEA Safety Guide, sites are categorized into three types based on S wave velocity (Type 1: *Vs* > 1100 m/s; Type 2: 1100 m/s > *Vs* > 300 m/s; Type 3: 300 m/s > *Vs). * Where *Vs* is the best estimate shear (S) wave velocity of the foundation medium just below the foundation level of the structure in the natural condition for very small strains. Shear (S) wave velocity measurements were carried out in 4 boreholes by means of the acoustic wave logging method (see basic material [L. 165]) and with the aid of the GOI logging instrument and the ALT FWS40 probe. It should be mentioned that initial measurements of the velocity of seismic waves on the main construction site were executed in 1980 (Minenergo Moscow) and 1992 [L.

103]. New measurement results of the Vs shear wave velocity are indicated in Tab. 113 below.

Tab. 113 Shear wave (S) velocity values for individual rock types and weathering degree (according to basic material [L. 165]).

Rock	Vs ($m.s^{-1}$)
Biotite paragneiss - fresh	1845 – 1971
Biotite paragneiss - slightly weathered	1858 – 1963
Biotite paragneiss - moderately weathered	1837 – 1956
Biotite paragneiss - highly weathered	1559 – 1977
Biotite paragneiss - completely weathered	1597 – 1970
Vein granite to pegmatite - fresh	1843 – 2266
Pegmatite - fresh to slightly weathered	1924 – 1979
Pegmatite - moderately weathered	1763 – 1911

The data shown in the table above imply that the shear (S) wave velocity on the construction site of Temelín NPP safely exceeds the limit value for type 1 sites, i.e. 1100 $m.s^{-1}$. The Temelín NPP construction site is thus categorized as type 1 site in accordance with Paragraph 3.1 of the IAEA Safety Guide NS-G-3.6 [L. 13].

2.6.2.3.8 Hydrogeological Conditions on the Construction Site and the Site Vicinity

The site vicinity and the ETE3,4 construction site are situated in an area that is rated as a hydrogeologically less significant structure. Crystalline rocks form a rock complex of very low permeability with relatively higher permeability of the weathering mantle in the subsurface fissure disintegration zone, in tectonically disrupted zones, and in more rigid rock inserts. In these zones, a network of open fissures is created, which facilitates the flowing of groundwater and the formation of small-sized groundwater aquifers.

Two mutually independent and spatially separated groundwater horizons may be found on the ETE3,4 construction site and the site vicinity:

- Shallow aquifer bound to Quaternary sediments and the near-surface eluvium zone, mostly along the boundary of Quaternary sediments and eluviums, or on the eluvium base and in the subsurface fissure disintegration zone;
- Deep aquifer bound to the fractire system of the deeper bedrock

Shallow Groundwater Horizon

The weathering mantle and the Quaternary cover along with the zone of surface rock disintegration of the bedrock create a uniform aquifer with a shallow circuit of intrinsic-fissure permeability, which changes to fissure permeability with increasing depth.

Quaternary sediments and eluvial zones are saturated discontinuously, often only temporarily. Their permeability is low. This aquifer with a groundwater level under slight pressure and occasionally markedly fluctuating groundwater table is significantly influenced by climactic factors, since the groundwaters are complemented by infiltration from atmospheric precipitation in the whole territory.

The filtration coefficient determined by means of inflow tests carried out in "RK" series boreholes in the site area of Temelín NPP ranged from 10^{-6} to 10^{-7} $m.s^{-1}$ [L. 51]. The examined layer comprised weathered crystalline rocks, their eluviums, backfills and the preserved Quaternary cover.

The groundwater flow in this horizon is directionally dependent on the terrain morphology. With a view to the situation of the site in the top section of the old peneplain, groundwater flows in all directions with respect to the local base levels. The final base-level is the valley of the Vltava River.

The shallow circulation of groundwater manifests to depths between 25 and 40 metres. The yield of potential discharges in the drainage area amount to tenths and more often to hundredths of a litre per second (see basic material, e.g. [L. 113]).

The shallow aquifer was substantially disrupted by the construction of Temelín NPP with the installed capacity of 2 x 1000 MW. It is very likely that its regime within the site area and its vicinity currently depends especially on the surface water draining system and on the circulation conditions in man made grounds and backfill soils. The groundwater level on the ETE3,4 construction site is primarily affected by the previously performed rough ground shaping of the terrain and excavation works.

It is thus possible to state that the precipitation-runoff conditions on the ETE3,4 construction site are complicated. Nevertheless, these situation may be described thanks to the good knowledge of buried structures, pipelines and other construction interventions in the land.

In terms of chemical composition, the waters are of medium total mineralisation, neutral to weakly acid, with a predominant presence of HCO_3 - SO_4 - Ca - Mg ions. According to CSN EN 206-1 (Tab. 132), they mostly represent a slightly aggressive environment in relation to the content of aggressive CO_2 (locally also medium aggressive – see e.g. basic material [L. 155], [L. 156]) and a slightly aggressive environment in relation to the content of sulphate ions (cf. [L. 165]).

Fracture Groundwater Horizon

The groundwater regime of the deeper lying aquifer in depths from 50 to more than 100 metres is characterised by stagnating or slow-flowing groundwater. A very long retention period is anticipated in relation to the groundwater of this aquifer, as documented by the determined age (10,000 years and more, see basic material [L. 184]).

As the groundwater of this aquifer do not communicate with shallow or surface waters, it is presumed that there is any potential effect of the siting of ETE3,4 on this type of groundwater.ETE3,4

2.6.3 REQUIREMENTS AND CRITERIA

The requirements and criteria relating to the content of Section 2.6 are divided into four segments in compliance with the structure of the IAEA Safety Standard NS-R-3 [L. 6]). Exclusion criteria and requirements are marked in bold letters.

The interpretation of the criteria pursuant to Decree No. 215/1997 Coll. [L. 1], evaluation methods and the formulation of the individual criteria demonstrations are specified in basic material [L. 166]). Within the interpretation of the criteria, allowance was also made to the wording of the SÚJB BN-JB-1.14 Safety Guide [L. 268]

Tab. 114 Hazards associated with earthquake occurrence

ID	Section	Criterion requirement according to Decree No. 215/1997 Coll.	Paragraph	IAEA Safety Requirements No. NS-R-3
7.1			3.1	The seismological and geological conditions in the region and the engineering geological aspects and geotechnical aspects of the proposed site area shall be evaluated.
			3.2	Information on prehistorical, historical and instrumentally recorded earthquakes in the region shall be collected and documented.
			3.3	The hazards associated with earthquakes shall be determined by means of seismotectonic evaluation of the region with the use of the greatest possible extent of the information collected.
			3.4	Hazards due to earthquake induced ground motion shall be assessed for the site with account taken of the seismotectonic characteristics of the region and specific site conditions. A thorough uncertainty analysis shall be performed as part of the evaluation of seismic hazards.
	Section 4e)	The achievement or exceeding of the value of intensity of the maximum design earthquake 8° MSK-64 (the Medvedev-Sponheuer-Karnik scale for the rating of macroseismic effects of earthquakes) on the lands of presumed siting.		
	Section 5c)	The achievement of the value of intensity of the maximum design earthquake in the range from 7° to 8° MSK-64.		

See also Article 5.2 of Annex E to WENRA Reactor Safety Reference Levels - January 2008 [L. 27]: "Site parameters to be considered in a nuclear power plant design - bullet 5 - earthquakes".

Tab. 115 Hazards arising from tectonic activity - potential for surface faulting in the territory

ID	Section	Criterion requirement according to Decree No. 215/1997 Coll.	Paragraph	IAEA Safety Requirements No. NS-R-3
7.2	Section 4f)	The occurrence of capable and seismically active faults with concurrent surface deformations in the area and the possibility of secondary fault formation, found by a geological survey on the land of presumed siting.	3.5	The potential for surface faulting (i.e. fault capability) shall be assessed for the site. The methods to be used and the investigations to be made shall be sufficiently detailed so that a reasonable decision can be reached using the definition of fault capability given in paragraph 3.6.
			3.6	A fault shall be considered capable if, on the basis of geological, geophysical, or seismological data, one or more of the following conditions applies: a) It shows evidence of past movement or movements (significant deformations and/or dislocations) of a recurring nature within such a period that it is reasonable to infer that further movements at or near the surface could occur. In highly active areas, where both earthquake data and geological data consistently reveal short earthquake recurrence intervals, periods of the order of tens of thousands of years may be appropriate for the assessment of capable faults. In less active areas, it is likely that much longer periods may be required. b) A structural relationship with a known capable fault has been demonstrated such that movement of the one may cause movement of the other at or near the surface. c) The maximum potential earthquake associated with a seismogenic structure is sufficiently large and at such a depth that it is reasonable to infer that, in the geodynamic setting of the site, movement at or near the surface could occur.

ID	Section	Criterion requirement according to Decree No. 215/1997 Coll.	Paragraph	IAEA Safety Requirements No. NS-R-3
			3.7	Where reliable evidence shows the existence of a capable fault that has the potential to affect the safety of the nuclear installation, an alternative site shall be considered.

ID	Section	Criterion requirement according to Decree No. 215/1997 Coll.	Paragraph	IAEA Safety Requirements No. NS-R-3
7.3	Section 4i	The occurrence of tectonic activity in the site vicinity zone which, in the period of operation of the facility or work site, would lead to a change of the present surface slope of the lands selected for the siting in an extent exceeding the stipulated technological requirements.		

Tab. 116 Geological and geotechnical hazards

<table>
<tr><th>ID</th><th>Section</th><th>Criterion requirement according to Decree No. 215/1997 Coll.</th><th>Paragraph</th><th>IAEA Safety Requirements No. NS-R-3</th></tr>
<tr><td rowspan="3">7.4</td><td>Section 4g)</td><td>The occurrence of geodynamic phenomena, such as landslides, block slides, plastic uplift of the subsurface and soil liquefaction, which endanger the rock massif stability on the land selected for the siting.</td><td rowspan="2">3.33</td><td rowspan="2">The site and its vicinity shall be evaluated to determine the potential for slope instability (such as land and rock slides and snow avalanches) that could affect the safety of the nuclear installation.</td></tr>
<tr><td rowspan="2">Section 5a)</td><td rowspan="2">Other karstic phenomena that are not specified in Section 4(c) of this decree and active geodynamic phenomena in the localities selected for the siting.</td></tr>
<tr><td>3.34</td><td>If there is found to be a potential for slope instability that could affect the safety of the nuclear installation, the hazard shall be evaluated by using parameters and values for the site specific ground motion.</td></tr>
<tr><td>7.5</td><td>Section 4c)</td><td>The occurrence of karstic phenomena in an extent endangering the stability of the rock massif in the subsurface and in the overlying stratum of the lands or area selected for the siting.</td><td>3.35</td><td>Geological maps and other appropriate information for the region shall be examined for the existence of natural features such as caverns, karstic formations and human made features, such as mines, water wells and oil wells. The</td></tr>
</table>

ID	Section	Criterion requirement according to Decree No. 215/1997 Coll.	Paragraph	IAEA Safety Requirements No. NS-R-3
	Section 5a)	Other karstic phenomena that are not specified in Section 4(c) of this decree and active geodynamic phenomena in the localities selected for the siting.		potential for collapse, subsidence or uplift of the site surface shall be evaluated.
	Section 4h)	The occurrence of the present or expected surface deformations of the area selected for the siting and their site vicinity zones as the consequence of exploitation of gas, oil, water or deep mining of minerals, the application of technologies for dissolving (leaching) minerals and their withdrawal, which may endanger the rock massif stability in the subsurface, or the overlying stratum of the structure.	3.36	If the evaluation shows that there is a potential for collapse, subsidence or uplift of the surface that could affect the safety of the nuclear installation, practicable engineering solutions shall be provided or otherwise the site shall be deemed unsuitable.
	Section 4n)	The occurrence of old mining activities in the site vicinity zones where there is a risk of undermining consequences, pit water bursts, and of the destruction effects of extensive mountain or pressure bumps.	3.37	If there do seem to be practicable engineering solutions available, a detailed description of subsurface conditions obtained by reliable methods of investigation shall be developed for the purpose of determination of the hazards.
	Section 4o)	The occurrence of raw material mining in the site vicinity zones, which could have an unfavourable impact on the construction and the operation of the facility or the work site.		
7.6	Section 4d)	The manifestations of post-volcanic activity, such as the emanation of gases, thermal, mineral and mineralised waters, found on the lands or area of the presumed siting and in their site vicinity zones.	3.52	Historical data concerning phenomena that have the potential to give rise to adverse effects on the safety of the nuclear installation, such as volcanism, sand storms, severe precipitation, snow, ice, hail, and subsurface freezing of subcooled water (frazil), shall be collected and assessed. If the potential is confirmed, the hazard shall be assessed and design bases for these events shall be derived.
7.7	Section 4k)	The bearing capacity of the foundation soils on the lands selected for the siting lower than 0.2 MPa and with foundation soils of a collapsible and very expansive nature or with a more than 3% share of organic components and layer thickness inhibiting their removal or replacement.	3.41	The geotechnical characteristics of the subsurface materials, including the uncertainties in them, shall be investigated and a soil profile for the site in a form suitable for design purposes shall be determined.
			3.42	The stability of the foundation material under static and dynamic

ID	Section	Criterion requirement according to Decree No. 215/1997 Coll.	Paragraph	IAEA Safety Requirements No. NS-R-3
	Section 5b)	Unfavourable properties of the foundation soils, surrounding soils and rocks on the lands selected for the siting.		(seismic) loading shall be assessed.
7.8	Section 4g)	The occurrence of geodynamic phenomena, such as landslides, block slides, plastic uplift of the subsurface and soil liquefaction, which endanger the rock massif stability on the land selected for the siting.	3.38	The potential for liquefaction of the subsurface materials of the proposed site shall be evaluated by using parameters and values for the site specific ground motion.
			3.39	The evaluation shall include the use of accepted methods of soil investigation and analytical methods to determine the hazards.
			3.40	If the potential for soil liquefaction is found to be unacceptable, the site shall be deemed unsuitable unless practicable engineering solutions are demonstrated to be available.
7.9	Section 4m)	In the area of underground structures, the impossibility of covering the main part of the underground structure by a rock massif with thickness larger than three maximum widths of the underground structure, however, not less than 30 metres.		
	Section 5g)	High interstitial or fissure permeability of the rocks identified by the geological survey of the underground structures.[34]		
	Section 4l)	t The occurrence of geological conditions in the area selected for the siting, such as water-bearing soils, non-cohesive or soft cohesive soils, which predetermine the 3rd degree of the tunnel structure driving.[35]		
	Section 5h)	The occurrence of geological conditions, which predetermine the 2nd degree of tunnel structure driving[36] of underground structures.		

[34] CSN 75 1400 Surface Water Hydrological Data, 1990

[35] Technical standard: UN 73 7010 Tunnels and Other Subsurface Structures

[36] Section 8 of Act No. 266/1994 Coll., on Railways

Tab. 117 Hydrogeological risks and the hydrogeological aspects of the environmental impact of the installation

<table>
<tr><th>ID</th><th>Section</th><th>Criterion requirement according to Decree No. 215/1997 Coll.</th><th>Paragraph</th><th>IAEA Safety Requirements No. NS-R-3</th></tr>
<tr><td>7.10</td><td>Section 4j)</td><td>The existence of significant supplies of groundwaters or mineral waters in the site vicinity zones, in which permanent depreciating water changes would occur as a result of the construction or operation of construction project in terms of radiation influence.</td><td>4.7</td><td>A description of the groundwater hydrology of the region shall be developed, including descriptions of the main characteristics of the water bearing formations, their interaction with surface waters and data on the uses of groundwater in the region.</td></tr>
<tr><td rowspan="4">7.11</td><td rowspan="2">Section 5d)</td><td rowspan="2">The occurrence of hydrogeological conditions on the construction plots, which complicate the monitoring and prediction of groundwater behaviour.</td><td>3.43</td><td>The groundwater regime and the chemical properties of the groundwater shall be studied.</td></tr>
<tr><td rowspan="2">4.8</td><td rowspan="2">A programme of hydrogeological investigations shall be carried out to permit the evaluation of radionuclide movement in hydrogeological units. This programme should include investigations of the migration and retention characteristics of the soils, the dilution and dispersion characteristics of the aquifers, and the physical and physicochemical properties of underground materials, mainly related to transfer mechanisms of radionuclides in groundwater and their exposure pathways.</td></tr>
<tr><td rowspan="2">Section 5f)</td><td rowspan="2">The occurrence of well permeable soils and groundwater levels in the depth of less than 2 metres under the planned level of rough ground shaping of the lands selected for the siting,</td></tr>
<tr><td>4.9</td><td>Assessment of the potential impact of the contamination of groundwater on the population shall be performed by using the data and information collected in a suitable model.</td></tr>
</table>

ID	Section	Criterion requirement according to Decree No. 215/1997 Coll.	Paragraph	IAEA Safety Requirements No. NS-R-3
7.12	Section 5e)	The occurrence of aggressive groundwater with potential contact with the construction structures on the lands selected for the siting.	3.43	The groundwater regime and the chemical properties of the groundwater shall be studied.

2.6.4 BASIC MATERIALS AND DATA USED WITHIN THE ASSESSMENT

The assessment of geological and seismological hazards was performed with the aid of numerous basic materials and data pertaining to the results of geological, seismological, and engineering geological surveys of the individual territorial units of the Temelín NPP region.

When assessing the hazards associated with the occurrence of an earthquake in the region of Temelín NPP, the information presented in the Pre-Operational Safety Report (POSR) of Temelín NPP [L. 18] was utilised. The report contains a description of Temelín NPP, its geological structure, seismotectonic model, as well as a detailed analysis of the procedures applied within the determination of the SL-2 value. In addition, basic materials and studies elaborated as a part of periodical reviews of the POSR of Temelín NPP [L. 160] were used during the evaluation of these hazards as underlying materials for the determination of the seismic hazard of Temelín NPP [L. 161] and [L. 162].

Namely the report on the supplementary survey of the near region of Temelín NPP performed in 1995 [L. 187] served as the basis for the evaluation of the movement potential of the faults in the near region of Temelín NPP. The report summarizes findings on the geology, geomorphology and tectonics of the near region of Temelín NPP as recorded in an abundant set of research reports on the geological mapping of the region of Southern Bohemia, as well as of numerous construction geological, hydrogeological and raw deposit expert opinions. Within the framework of supplementary surveys, exploratory activities were performed - i.e. geophysical measurements, geological mapping, and exploratory drilling. The results of the supplementary survey were also included, in concise form, in the POSR of Temelín NPP [L. 18]. Furthermore, basic materials and studies processed as a part of periodical reviews of the POSR of Temelín NPP [L. 158], [L. 190] along with findings from newly executed projects were utilised for the evaluation: "Paleoseismological Evaluation of the Survey of Fault Structures in the Surroundings of Temelín NPP" carried out by a group of authors from the Institute of Physics of the Earth of Masaryk University Brno, Energoprůzkum Praha and the Institute of Rock Structure and Mechanics of the Academy of Sciences of the Czech Republic [L. 188]) and "Verification of Movement Activity of N-S Faults in the Temelín NPP" near region carried out by Energoprůzkum Praha [L. 164] and [L. 167].

The assessment of engineering and geological hazards was performed based on a number of basic materials containing the results of engineering geological surveys of the main construction site of the existing Temelín NPP, geological maps and databases kept by Geofond Prague, as well as with the aid of other specialised studies indicated in the following overview:

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2.6.5 METHODS USED WITHIN THE ASSESSMENT

With a view to the extensiveness of the issue assessed in Chapter 2.6, the use of several methods was required depending on the concrete evaluated phenomenon. Some of the more general methods that were used within the assessment are detailed in the following text, and descriptions of specialised procedures are included in the subsections, in which the individual requirements and criteria are evaluated. The general assessment approach according to the criteria stipulated in Decree No. 215/1997 Coll. [L. 1], including the interpretation of the individual criteria and the determination of the scope of their proof, is described in [L. 166].

2.6.5.1 METHODS USED WITHIN SEISMIC HAZARD ASSESSMENT

The procedures recommended for the delimitation of the region, as well as the focus of survey activities and evaluations are defined in Paragraphs 3.7 to 3.10 of the IAEA Specific Safety Guide No. SSG-9 [L. 14], and the procedures involving the development of a seismological database and its recommended content are laid down in Paragraphs 3.24 to 3.33 thereof. In addition, seismological procedures were applied that are typically used, for example, during the compilation of earthquake catalogues, evaluations of recorded events, verifications of catalogue completeness (historical seismology), etc., or within the registration of instrumental records.

Seismological procedures also include methods for obtaining data on site-specific seismic energy attenuation in relation to the distance from the source, as well as procedures implemented when developing attenuation formulas or evaluating uncertainties.

Another group of methods comprises mathematical procedures implemented in connection with the determination of SL-2. These methods have experienced dynamic development over the past few years, especially in terms of probabilistic approaches (see e.g. lit. [L. 197]), and they are gradually introduced in IAEA recommendations and guides (e.g. Sections 6 and 7 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]).

Assessments of the seismic hazard of Temelín NPP were always performed with the aid of the most current and state-of-the-art seismological methods at the time of the particular assessment. For example, deterministic and probabilistic approaches in compliance with the recommendations stipulated in the then current IAEA Safety Guides Nos. 50-SG-S1 (1991) and NS-G-3.3 (2002) were applied in relation to the methods used for the determination of SSE (Safe Shutdown Earthquake). A detailed description of the procedures is provided in the POSR of Temelín NPP (basic material [L. 18]).

The subsequent gradual revalidations of the seismic hazard of Temelín NPP, included in the individual POSR versions, did not differ substantially from one another and they always confirmed the validity of the seismic design value of Temelín NPP (in accordance with the valid IAEA Safety Guides, the ground acceleration value equal to 0.1 g was adopted as SL-2 for NPP1,2).

In connection with the ongoing development of the methods and mathematical procedures in the area of seismic hazard assessment and also in response to the publication of the amended IAEA Specific Safety Guide No. SSG-9 [L. 14] in 08/2010, a new SL-2 calculation method based on a probabilistic approach was developed (see basic material [L. 131]) and subsequently used for SL-2 calculations (see basic material [L. 132]).

The assessment carried out within the Initial Safety Analysis Report, the main purpose of which was to decide whether seismicity was an exclusion criterion for the siting of ETE3,4 in the Temelín NPP site pursuant to Section 4(e) of Decree No. 215/1997 Coll. [L. 1], SSE indicated in the POSR of Temelín NPP was used as a decisive value [L. 18]. The result of the latest seismic hazard revalidation is also provided, i.e. SL-2 is expressed by the acceleration of ground motion (see basic material [L. 132]).

2.6.5.2 METHODS FOR ASSESSING THE HAZARDS ASSOCIATED WITH FAULT MOVEMENTS

During the evaluation of the above indicated requirements and criteria, the initial focus was on the question of whether, based on the findings mentioned above, a fault with potential for displacement at or near the ground surface was known to be present on the land of presumed siting or in the site vicinity, or whether the existence of such faulting was suspected (see Paragraph 8.6 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]: "When faulting is known or suspected to be present...").

The "suspicion" was namely supported by the analysis of the geological and geomorphological characteristics of the territory, which are significant for the formulation of such "suspicion" and the age of which is relevant in relation to the type of the local tectonic environment (see Paragraph 3.14 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]). For intraplate regions, Paragraph 3.12 of the IAEA Specific Safety Guide No. SSG-9 recommends evaluating tectonic information from the period of the Pliocene to the present. The basic geological and geomorphological characteristics of the territory that may support the "suspicion" of the existence of surface faulting include:

- Occurrence of linear topographic or structural relief elements (fault slopes, straight running slopes, lineaments);
- Occurrence of distinct lithological boundaries;
- Occurrence of rock mass indicating mechanical rock transformation along tectonic lines or the occurrence of clay and other minerals formed in near-surface conditions;
- Occurrence of micro-earthquakes or local historical earthquakes, especially in geographic coincidence with the specified characteristics.

The conditions under which the identified fault is considered as capable of displacement (surface deformation) are defined in Paragraph 8.4 of the IAEA Specific Safety Guide No. SSG-9 [L. 14] and they are identical to the conditions stipulated in Paragraph 3.6 of the IAEA Safety Requirements No. NS-R-3 [L. 6] (see the requirements and criteria in the table above).

When determining potential fault capability, consideration should also be given to the possibility that faults that have not demonstrated recent near surface movement may be reactivated by reservoir loading, fluid injection in the fault structure, water or fluid withdrawal from the structure, etc. (see Paragraph 8.7 of the IAEA Specific Safety Guide No.SSG-9 [L. 14]).

On the other hand, fault creep evaluation is categorized outside the scope of application of the IAEA Specific Safety Guide No. SSG-9.

During field research of tectonic history of faults, a wide range of research methods was implemented, such as geological map data analyses, geomorphological analyses, geophysical property evaluations, geological mapping, river terrace occurrence evaluations, archive drill evaluations, exploratory drilling, exploratory trenching, rock sample dating, micro-earthquake occurrence analyses, etc. Similarly to the procedures applied within the resolution of the subject issues, these methods are described in the relevant survey reports (e.g. basic materials [L. 188] and [L. 187])

Paleoseismological methods and procedures were newly applied within the investigations of the near region of ETE3,4, namely with the aim of verifying the presumptions regarding the latest movements of faults and to possibly add data on prehistoric earthquakes to the seismological database.

Finally, it should be noted that the question of potential for fault capability was considered with reference to the exclusion criteria stipulated in Section 4(f) and (i) of Decree No. 215/1997 Coll. [L. 1], i.e. for the site vicinity and the construction site of ETE3,4. The results of the assessment of the faults present in the near region, especially in connection with the occurrence of potential prehistoric earthquakes, will be included in the subsequent ETE3,4 safety documentation phase.

2.6.5.3 METHODS FOR ASSESSING ENGINEERING AND GEOLOGICAL HAZARDS

A wide variety of methods and procedures was used within the assessment of engineering and geological hazards, ranging from evaluations of available map data to specialised geotechnical survey methods. These methods and procedures are described in detail in the used basic materials (e.g. [L. 159], [L. 144] to [L. 150]). During the latest surveys [L. 165], the investigation procedures were complemented with the selected methods recommended by the IAEA Safety Guide No. NS-G-3.6 [L. 13].

When evaluating the requirements and criteria specified in Tab. 116 under Sections 7.4, 7.7 and 7.8, rock and soil parameters and other subsoil characteristics defined in [L. 144] to [L. 150], [L. 155], [L. 156] and [L. 165] were implemented.

The evaluation of the requirements and criteria provided in Tab. 116 under Sections 7.5 and 7.6 is based on a comparison of the information obtained from maps [L. 72] to [L. 76] and other basic materials with the wording of the individual criteria, and on a decision on the occurrence of the relevant geological phenomenon in the area delimited by the criterion. In relation to the phenomena, the occurrence of which had been confirmed in the subject area, an assessment of the phenomenon (occurrence) severity in terms of ETE3,4 nuclear safety was executed.

Hydrogeological hazards (see Tab. 117) were evaluated with the aid of data contained in the WRI database (Hydroecological Information System, T. G. Masaryk Water Research Institute) and expert opinions and reports on the results of engineering geological surveys of the Temelín NPP construction site (see basic material [L. 51], [L. 107], [L. 113], [L. 155], [L. 156], [L. 165] and [L. 184]. As regards the phenomena, the occurrence of which had been confirmed in the subject area, an assessment of the phenomenon (occurrence) severity in terms of ETE3,4 nuclear safety was executed and an investigation was carried out to ascertain the existence of engineering measures that could overcome or limit such phenomena influence.

2.6.6 DELIMITATION OF THE EVALUATED TERRITORY

The size of the territorial units that were subjected to the evaluation was determined in a manner that would ensure that the evaluation covered an area where the occurrence of any event or phenomenon could affect the safety of the nuclear installation.

The procedures and the reasons for the given delimitation of the territory where the evaluation took place are explained in the comments to the individual selection criteria, unless the text of a particular requirement or criterion distinctly defines the territory to be subjected to the evaluation.

In connection with the determination of the size of the territorial units, we primarily start from Decree No. 215/1997 Coll. [L. 1], which defines the terms "locality", "site vicinity zone" and "lands of supposed siting", whereas:

- Locality shall mean an area within a distance of 20 km from the boundary of the land proposed for the siting;
- Site vicinity shall mean an area within a distance of 3 km from the boundary of the land proposed for the siting;
- The land of supposed siting is not specifically defined by the Decree, however, for the purposes hereof, we have associated it with the construction site of ETE3,4, i.e. with the territory where crucial power plant structures are to be placed.

Furthermore, the recommendations presented in the IAEA Specific Safety Guide No. SSG-9 [L. 14] were also considered when determining the size of the evaluated territorial units. According to the IAEA Specific Safety Guide, the following territorial units with their Czech equivalents are delimited for the purpose of assessing the seismic hazard of structures with nuclear installations and the geotechnical properties of foundation soils on their construction sites:

Region	-	Region;
Near region	-	Lokalita (Locality);
Site vicinity	-	Užší lokalita (Close locality);
Site area	-	Areál jaderného zařízení (Nuclear facility grounds).

Pursuant to Paragraph 3.11 of the IAEA Specific Safety Guide No. SSG-9 [L. 14], a near regional study should cover an area typically "not less than 25 km in radius" from the nuclear installation. Site vicinity studies should cover an area typically "not less than 5 km in radius" from the nuclear installation (see Paragraph 3.16 of the IAEA Specific Safety Guide). The site area is defined in Paragraph 3.19 of the IAEA Specific Safety Guide No. SSG-9 [L. 14] as an area of one square kilometre, meaning an area on which the structures of the nuclear power plant are situated.

Within the evaluation, the western part of the main construction site of Temelín NPP [37] was considered as the ETE3,4 construction site, including the area contemplated for the siting of ETE3,4 units, which is outside the guarded area of Temelín NPP (areas S1 and S2 as shown in Fig. 25).

The delimitation of the site vicinity respects the wording of Decree No. 215/1997 Coll. [L. 1], which defines the site vicinity as an area within a distance of 3 km from the boundary of the land proposed for the siting. The reason for this selection arises from the finding that none of the evaluated engineering and geological phenomena, the occurrence of which has been confirmed in an area within a radius of more than 3 km

[37] The main construction site is the area investigated within engineering geological surveys conducted in relation to the siting and construction of Temelín NPP consisting of 4 VVER 1000 units. At present, it embraces the whole guarded area of the current Temelín NPP, including the western foreland (outside the existing guarded area), which is designated for the placement of the cooling towers of the new nuclear installation at Temelín NPP.

from the boundary of the construction site, is capable of jeopardizing the nuclear safety of NPP 3,4.

Pursuant to Paragraph 3.11 of the IAEA Specific Safety Guide No. SSG-9 [L. 14], the area within a radius of 25 km from the boundary of the land proposed for the siting of the nuclear installation was considered as the near region for evaluation purposes.

The above delimited territorial units are too small for a seismic hazard assessment. Evaluations of the occurrence of earthquakes and their source areas should be performed in the context of larger territorial units – i.e. a region, which the IAEA standards (see Paragraph 3.7 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]) define as an area with a radial extent of 300 km from the nuclear installation. The shape of the region may be assymetric and adjusted to include significant seismogenic sources.

Since the typical size of the region as recommended by the IAEA Safety Guides has experienced a considerable shift (from 150 km and more pursuant to Paragraph 3.5 of the IAEA Safety Guide No. NS-G-3.3 [L. 11] to 300 km according to the IAEA Specific Safety Guide No. SSG-9 [L. 14]), it was necessary to verify the validity of the determined seismic hazard as presented in the POSR of Temelín NPP1,2 [L. 18], in particular whether the calculated SSE (SL-2) value includes significant seismogenic areas situated within a radius of 300 km from Temelín NPP. A review of the procedures used for the determination of SSE (SL-2) indicated in the POSR of Temelín NPP [L. 18] has shown that the source areas situated within this radius were included in the calculations and thus, the determined SSE (SL-2) value is valid also for the new delimitation of the region of Temelín NPP.

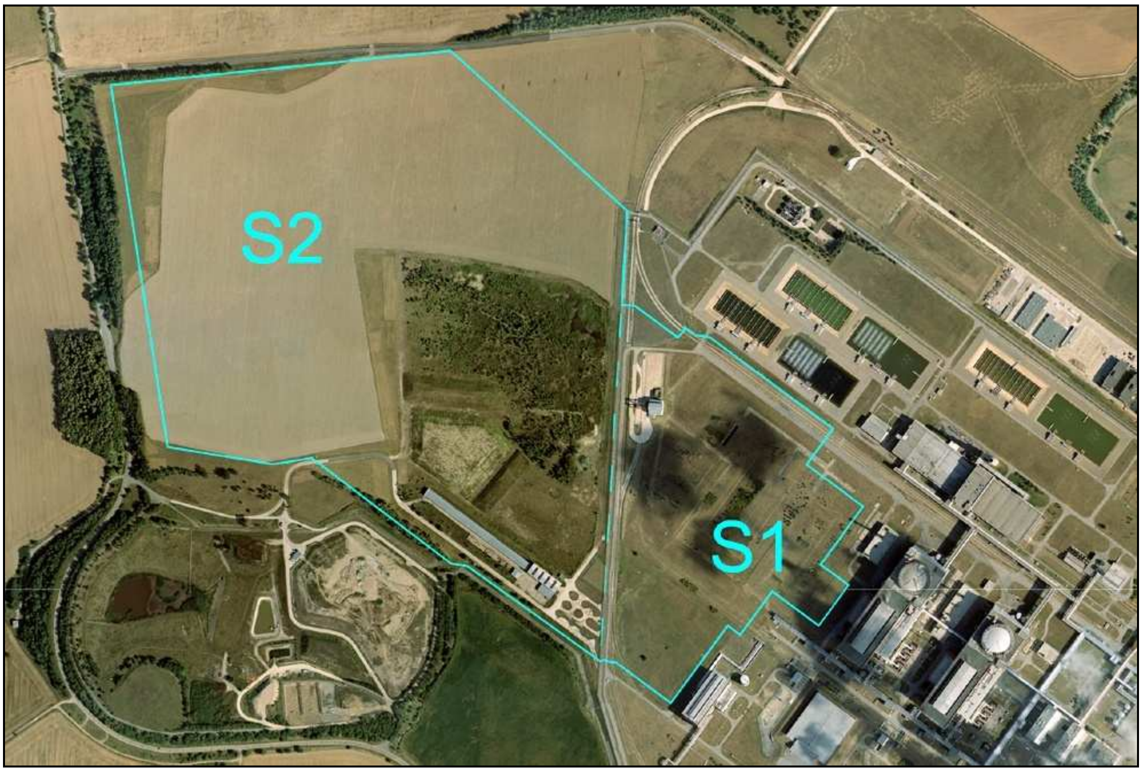

Fig. 25 Delimitation of the ETE3,4 construction site (plots S1 and S2).

In the course of the evaluations, the size of the region of Temelín NPP was enlarged to cover the radius of 300 km from the nuclear power plant in compliance with the recommendations of the IAEA Specific Safety Guide No. SSG-9 [L. 14], whereas the

Friuli source area was also included and focus was given to regional segments lying to the south and south-east of Temelín NPP. This preference is preconditioned by the location of the earthquake source areas that are decisive for the assessment of the seismic hazard of Temelín NPP, as well as by the generally known minor attenuation of Eastern Alp earthquakes propagating into the Bohemian Massif.

2.6.7 DETAILED EVALUATION OF ALL REQUIREMENTS AND CRITERIA ACCORDING TO DECREE NO. 215/1997 COLL. IN COMBINATION WITH REQUIREMENTS ACCORDING THE IAEA SAFETY STANDARD NO. NS-R-3

2.6.7.1 REQUIREMENTS ACCORDING TO SECTION 7.1

2.6.7.1.1 Specification of Hazards Grouped in Section 7.1

Section 7.1 of Tab. 114 groups together the hazards associated with the occurrence of earthquakes in the region of Temelín NPP.

In compliance with Decree No. 215/1997 Coll. [L. 1] (see Section 4(e)), the maximum permissible seismic load of the nuclear installation construction site is limited by the value of macroseismic intensity equal to "maximum calculated earthquake"[38] of ° MSK-64 scale. The conditioning criterion stipulated in Section 5(c) is conceived in a similar manner.

In seismological practice, the term "earthquake macroseismic intensity" is understood as the degree to which an earthquake affects the people, man-made structures and the landscape in a specific area. The numerical value of macroseismic intensity (I) is expressed by means of a macroseismic scale where each degree is characterised by a set of observable effects. In Europe, 12-degree macroseismic scales are used[39], including the already mentioned MSK-64 macroseismic scale (Medvedev-Sponheuer-Karnik scale).

Applying macroseismic intensity in practice, however, is problematic. The evaluations are rather subjective since everything depends on the observers and their estimation of the scope of damage. On the other hand, the determination of macroseismic intensity based on the effects of an earthquake on the landscape seems relatively objective. First attempts to quantify the macroseismic effects of an earthquake on the landscape appeared in connection with the development and application of paleoseismological research methods. The ESI-2007 may serve as an example [L. 136]).

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[38] The term "Maximum Calculated Earthquake" is the equivalent of the term "Safe Shutdown Earthquake" (SSE). For example, in manuals of the United States Nuclear Regulatory Commission (U.S. NRC), "Safe Shutdown Earthquake" is defined as the maximum potential earthquake with respect to region-specific geological and seismological conditions and site-specific subsurface properties. In terms of securing nuclear safety, certain nuclear installation systems, structures and components must be designed to remain functional even in case of maximum ground motion induced by such earthquake (see NRC regulations, Title 10, Chapter I, Part 100, Appendix A). "Safe Shutdown Earthquake" also means an earthquake, which a power plant is built to withstand under all design conditions (see European Utility Requirements for LWR Nuclear Power Plants, Revision C, April 2001, Appendix B, Definitions).

[39] At present, Europe uses the EMS-98, which is a detailed 12-degree macroseismic scale developed by an international work group of the European Seismological Commission.

Modern procedures, which determine seismic hazards with the aid of New Generation Attenuation (NGA) relationships, use magnitude (or moment magnitude [40] Mw) to measure the size of an earthquake and express its effect on the relevant site by the value of Peak Ground Acceleration (PGA).

Where acceleration is used as a referential value in the IAEA Safety Guides, SSE corresponds to SL-2 (Seismic Level 2), which is defined as "*the most stringent safety requirement*" (see Paragraph 9.1 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]). In practice, this level is associated with the size of ground motion that may occur with small probability. Some countries accept the probability that this value may be exceeded in the range from 1 x 10^{-3} to 1 x 10^{-4} (mean value) or from 1 x 10^{-4} to 1 x 10^{-5} (median) per year - see Paragraph 2.3 the IAEA Safety Guide No. NS-G-1.6; [L. 18 Předprovozní bezpečnostní zpráva pro 1. a 2. blok JE Temelín (Preoperational safety report for Temelín ETE1,2 Units)

L. 19].

The IAEA Safety Requirements No. NS-R-3 [L. 6] also stipulate the requirement to perform an evaluation of the seismic hazards when selecting the site of a nuclear installation. The approach to the evaluation of the effects of seismicity on the process of siting of a nuclear installation set out in the IAEA Specific Safety Guide No. SSG-9 [L. 14] differs from the approach defined in Decree No. 215/1997 Coll. [L. 1] Even if seismicity is not considered as an exclusion criterion[41], great emphasis is laid on the in-depth knowledge of site-specific seismological parameters. This approach also reflects in the requirements on the robustness of the designed seismic resistance of the structure[42] (see e.g. European Utility Requirements [L. 264]).

In general, the specific requirements concerning seismic evaluations are specified in Paragraphs 3.1 to 3.4 of the IAEA Safety Requirements No. NS-R-3 [L. 6], which also stipulate the following obligations:

3.1 The seismological and geological conditions in the region[43] and the engineering geological aspects and geotechnical aspects of the proposed site area shall be evaluated.

3.2 Information on prehistoric, historical and instrumentally recorded earthquakes in the region shall be collected and documented.

3.3 The hazards associated with earthquakes shall be determined by means of seismotectonic evaluation of the region with the use of the greatest possible extent of the information collected.

3.4 Hazards due to earthquake induced ground motion shall be assessed for the site with account taken of the seismotectonic characteristics of the region and specific site conditions. A thorough uncertainty analysis shall be performed as part of the evaluation of seismic hazards.

[40] Moment magnitude corresponds to the size of an earthquake expressed by the volume of released energy. Moment magnitude was introduced into practice by T. Hanks and H. Kanamori in 1979 (Journal of Geophysical Research, 84 (B5), pp. 2348–2350).

[41] In some IAEA states (including the Czech Republic), high seismic activity is applied as an exclusion criterion, typically from the level of 8° - 9° MSK-64 (cf. Paragraph 4.2(b) in TecDoc 1341).

[42] The issue of design robustness is not a part of the siting process. It is resolved in the design phase.

[43] The region of a nuclear installation is defined as an area with a typical radius of 300 km from the nuclear installation pursuant to Paragraph 3.7 of the IAEA Specific Safety Guide Nos. SSG-9.

2.6.7.1.2 Criteria According to Sections 4e and 5c (Decree No. 215/1997 Coll.)

Pursuant to Section 4(e) of Decree No. 215/1997 Coll. [L. 1], seismic load, which exceeds the Maximum Calculated Earthquake (MCE) value of macroseismic intensity equal to I = 8° MSK-64, is an exclusion criterion. In accordance with Section 5(c) the achievement of the value of intensity of the Maximum Calculated Earthquake in the range from 7° to 8° MSK-64 is a conditional criterion.

With regard to the above criteria, helpful leads aiding the evaluation of the ETE3,4 construction site included the following:

- A map showing the distribution of seismic load in the territory of the Czech Republic, expressed in macroseismic intensity values (i.e. seismic zoning map);
- Defined Maximum Calculated Earthquake values for the Temelín NPP construction site, expressed in macroseismic intensity values.

Evaluation Based on Seismic Zoning Map Analyses

The seismic load of a major part of the territory of the Czech Republic is generated by the energy released by remote earthquakes, especially in the area of the Eastern Alps (see Fig. 26), which propagates into the Bohemian Massif with relatively low attenuation.

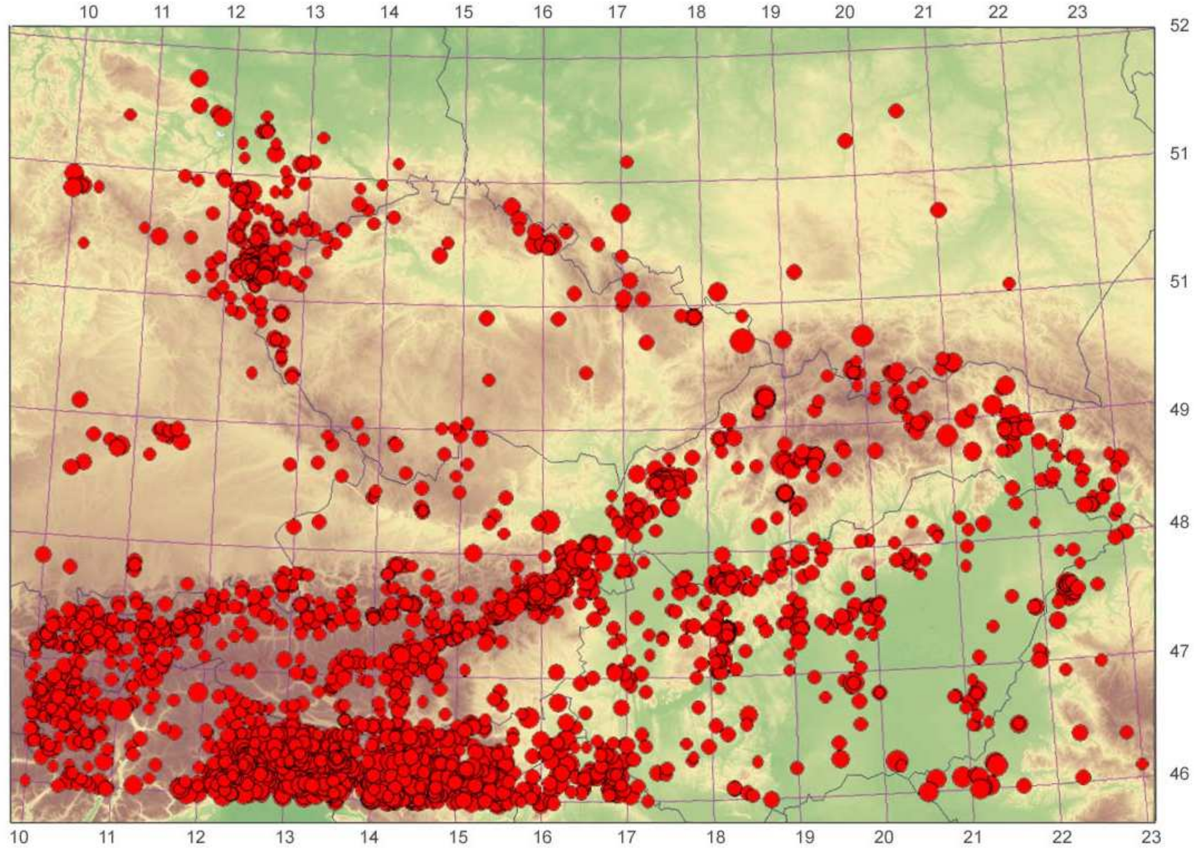

Fig. 26 Foci of earthquakes in Central Europe - 1200 - 2010; Mw ≥ 3.0. Data: Compilated catalogue of I. Prachař, version 2011 in: [L. 132].

In relation to the tectonic structures intersecting the territory of the Czech Republic, a higher seismicity level may be expected in the area of concentrated seismicity of Kraslice-Aš-Plauen in western Bohemia, in the surroundings of the Hronov-Poříčí

Fault in north-eastern Bohemia, in the surroundings of the Sudetic Marginal Fault at the boundary of north-eastern Bohemia and northern Moravia, and in the surroundings of the lineament of the Kopřivnice-Třinec elevation. The seismic potential in these areas is demonstrated by the occurrence of historical earthquakes with epicentre intensities ranging from 6.5° to 7.5° MSK-64 (see Tab. 118, Fig. 27).

Tab. 118 Overview of known strong historical earthquakes in source areas extending to the territory of the Czech Republic.

YYYY	MM	DD	HH	Mi	Lat.	Long.	Mw	I_0	Focus	Source area
1786	2	27	3	0	49.70	18.50	5.9	7.5	TOŠANOVICE	Area of the Kopřivnice-Třinec elevation
1785	8	22	7	0	49.70	19.00	4.4	6.5	BIELSKO-BIAŁA	
1901	1	10	2	30	50.50	16.10	5.1	7.0	HRONOV	Area of the Hronov-Poříčí Fault
1883	1	31	13	43	50.50	15.90	4.7	6.5	TRUTNOV	
1895	6	11	9	27	50.75	17.00	4.5	6.5	BRZEG	Area of the Sudetic Marginal Fault
1908	11	3	17	20	50.22	12.25	4.5	6.5	AŠ	Area of Kraslice-Aš-Plauen
1908	11	4	3	32	50.22	12.25	4.5	6.5	AŠ	
1908	11	4	13	10	50.22	12.25	4.5	6.5	AŠ	
1908	11	6	4	34	50.22	12.25	4.5	6.5	AŠ	
1985	12	21	10	16	50.22	12.50	4.4	6.5	NOVÝ KOSTEL	

Source: Compiled catalogue of historical Mw≥3.0 earthquakes of I. Prachař, version 2011 in: [L. 132].

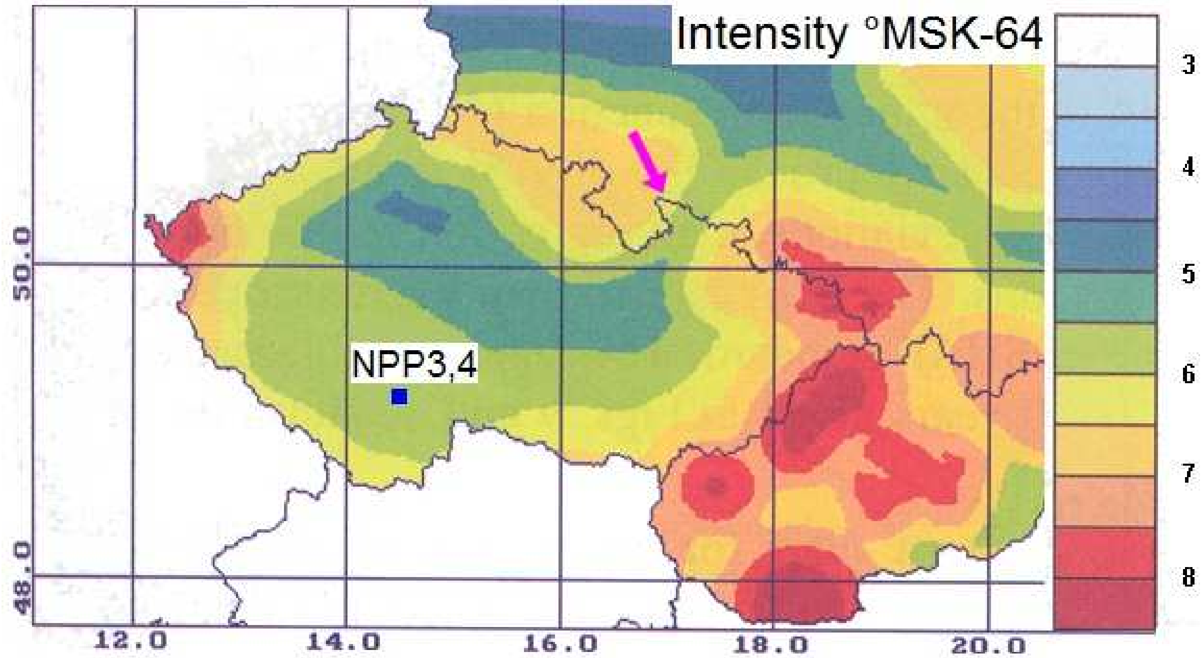

Fig. 27 Seismic load of the territory of the CR expressed in macroseismic intensity values for the return period of 5,000 years and with the 10% probability of exceedance within 527 years. Taken and modified from [L. 176]. The intensity values for the recurrence time of 10,000 years[44] are higher by approximately 0.5 of the intensity unit. The purple arrow points to an area near

44 In Czech seismological practice, the level of Maximum Calculated Earthquake as required by the criteria stipulated in Section 4e and 5c of Decree No. 215/1997 Coll. was associated with the presumption that the current seismic regime would be preserved in the future, i.e. over a horizon of 10,000 years. Some seismologists implementing the deterministic approach consider it as the maximum observed intensity value + 1°.

the Sudetic Marginal Fault where paleoseismological research has been completed recently and the results are not included in the map.

Evaluation Based on the MCE Determination for the Temelín NPP Construction Site

Determining MCE for the Temelín NPP site in tearms of macroseismic intensity was performed several times by different teams of authors between the years 1979 and 1998.

An older revision of the POSR of Temelín NPP contains an assessment of the seismic hazards executed with the aid of the seismostatistical approach. The authors[45] delimited the source zones based on [L. 171] and they determined macroseismic intensity attenuation in accordance with [L. 169]. The assessment included calculations of the curves of probability of the occurrence of earthquakes in the selected recurrence times (50 - 10,000 years), expressed in °MSK-64 macroseismic intensity according to [L. 117]. The maximum design values of macroseismic intensity for the Temelín Site were determined with the aid of the deterministic approach in the following manner:

6.5° MSK-64 in case of maximum shocks in the focal areas of Friuli and the Eastern Alps (for area delimitation see [L. 171]),

6° MSK-64 in case of maximum shocks in the focal areas of Regensburg – Augsburg and Innsbruck and their surroundings,

5.5° MSK-64 in case of maximum shocks in the focal area of Linz - Pregarten - Molln - Neulengbach. Another method used for MCE determination, which is also based on the deterministic approach, is the so-called seismotectonic method. It is founded on the interconnection of the foci of earthquakes with active fault segments (see lit. [L. 171]). The method is described in the older revision of the POSR of Temelín NPP [L. 18]). Its result is the determination of MCE in terms of macroseismic intensity provided that the current seismic regime remains preserved in the future, over a time horizon of 10,000 years. Maximum potential intensities were linked to the following faults:

6.5° MSK-64 in case of maximum earth shocks on the Mur-Mürz and Semmering lines,

6.0° MSK-64 in case of maximum earth shocks on the Leitha line (see basic material [L. 18]).

2.6.7.1.3 Requirements According to Paragraph 3.1 (NS-R-3)

As indicated in Paragraph 3.1 of the IAEA Specific Safety Guide No. SSG-9 [L. 14], an assessment of the hazards associated with the occurrence of earthquakes should be based on a comprehensive and integrated database of geological, geophysical, geotechnical and seismological information, as well as any other information that is relevant for the evaluation of the ground motion, tectonics and geological hazards at the site of construction of a nuclear installation (see Paragraph 3.2 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14]).

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[45] The author of the assessment was D. Procházková, its modified version was used by P. Šimůnek within the compilation of basic material for the IAEA mission in Temelín NPP in 2003.

Paragraph 3.3 of the IAEA Specific Safety Guide No. SSG-9 [L. 14] recommends to conduct investigations of the area of the nuclear installation on four spatial scales - regional, near regional, site vicinity and site area. The level of detail of the collected information and of the investigation work carried out in connection with the collection of such information should have an increasing tendency from the region towards the site.

In many areas, the information database requirements overlap and coincide with the requirements on the source data necessary for the fulfilment of other criteria and requirements, e.g. Section 4(f) and (i) of Decree No. 215/1997 Coll. [L. 1] and Paragraphs 3.5 to 3.7 and 3.42 of the IAEA Safety Requirements No. NS-R-3 L. 6].

For the Temelín NPP region, a database of seismological, geological and tectonic information has been developed and it is continuously complemented. The database includes, in particular:

- Geological and tectonic maps of the CR and other countries lying in the region of Temelín NPP or of larger geological units (Hercynides, Alpides, Carpathians);
- Seismic reflection profiles and their geological and tectonic interpretations;
- Satellite map of the region;
- Map of the MOHO discontinuity contours;
- Gravity map;
- Map of the current stress field.

In addition to graphic and map materials, the database also contains an abundant set of expert publications and articles. The latest review of the above specified findings on the region of Temelín NPP was performed in 2009 (see basic material [L. 160]) and complemented in 2011 (see basic material [L. 132]). A comprehensive description of the near region of Temelín NPP was elaborated in 2012 (see basic material [L. 168]).

The engineering and geological aspects of the Temelín NPP construction site, which relate to the seismological conditions on the construction site, were investigated during a supplementary survey of the site in 2010 (see basic material [L. 165]). It namely included the site characterization based on findings pertaining to S wave velocity in the vertical profile of the foundation soil and the identification of the foundation soil parameters under dynamic (seismic) load. The data are provided in Section 2.6.2.3.7.

2.6.7.1.4 Requirements According to Paragraph 3.2 (NS-R-3)

Paragraph 3.2 of the IAEA Safety Requirements No. NS-R-3 [L. 6] imposes an obligation to compile a composite catalogue of earthquakes that should include information on prehistoric, historical and instrumentally recorded earthquakes in the region of the nuclear installation. The catalogue of earthquakes is a significant part of the seismological database. It should extend as far back in time as possible (see Paragraph 3.25 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]. The scope of recommended information to be collected on each earthquake is defined in Paragraph 3.26 thereof for historical earthquakes and in Paragraph 3.27 thereof for instrumentally recorded earthquakes.

Data on Prehistoric Earthquakes

Records of prehistoric earthquakes have been so far obtained from three localities of the region of Temelín NPP. Apart from that, paleoseismological investigations were carried out in the near region of Temelín NPP, which included the uncovering of selected faults considered as suspicious (for details see basic materials [L. 188], [L. 167], [L. 168]). The results of these investigations are discussed in Section 2.6.7.2

The basic data on prehistoric earthquakes are presented in Tab. 119. Their position is shown in Fig. 28 where these indications are assigned to known seismogenic structures.

Tab. 119 Data on prehistoric earthquakes

No.	Near region	Lat.	Long.	Mw	Fault	Estimated time of last displacement
1	Municipality of Kopanina (western Bohemia)	50.204	12.461	6.3	Mariánské Lázně Fault	Approx. 4,000 years
2	Municipality of Bílá Voda (north-eastern Bohemia)	50.439	16.907	6.5	Sudetic Marginal Fault	Approx. 2,600 years
3	Siehdichfür Farm (north-eastern Austria)	48.291	16.678	7.0	Markgrafneusiedl Fault	16.2 thousand years*)

*) 5 events between 103.5 Ka and 16.2 thousand years [L. 78].

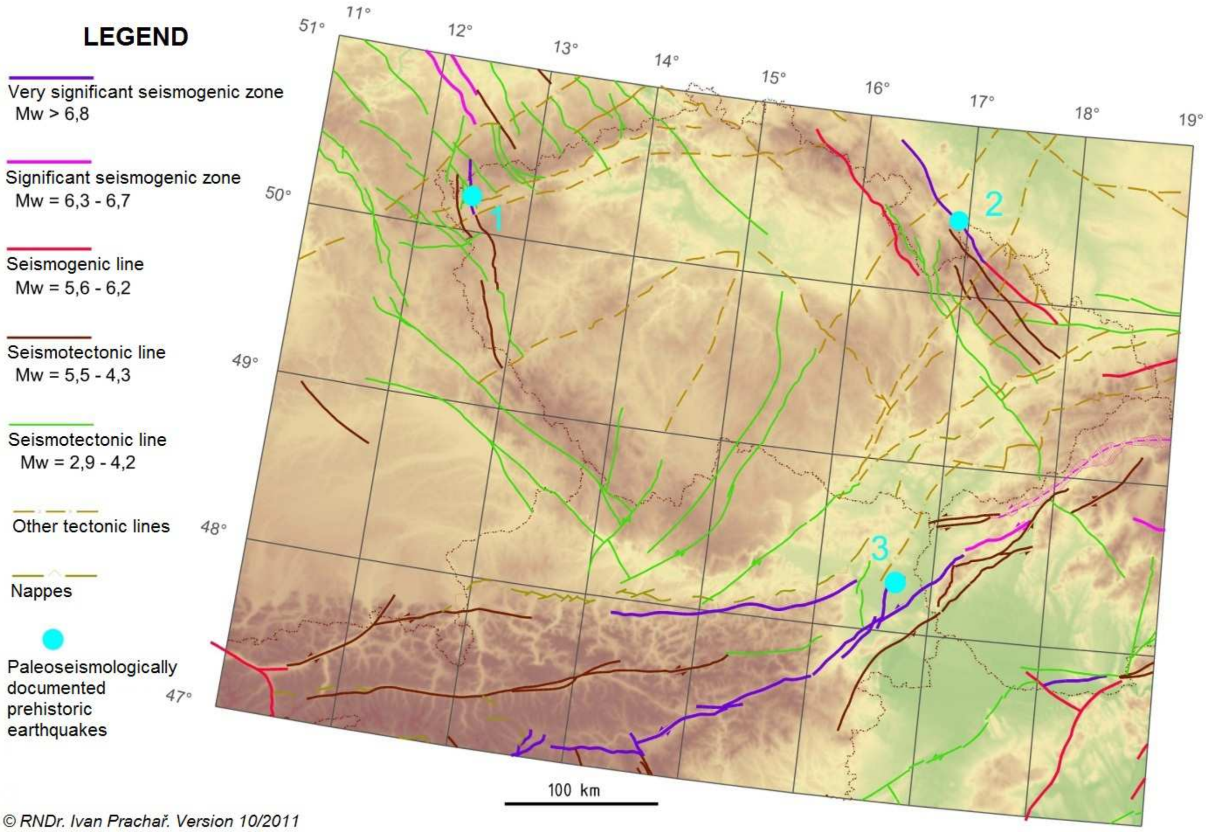

Fig. 28 Plotting of the positions of paleoseismological indications of prehistoric earthquakes. The numbers coincide with the numerical designation of localities in Tab. 119.

The data on prehistoric earthquakes Nos. 1 and 2 have not been published for the time being. They were obtained from oral information provided by P. Štěpančíková[46]. The data on the prehistoric earthquake in Austria were drawn from [L. 78].

Data on Historical Earthquakes

The data on historical earthquakes were also collected and processed in the form of a compiled catalogue of earthquakes (In: [L. 132]). The selected events occurring in the wider region of Temelín NPP (including instrumentally recorded) or strong earthquakes affecting the assessed SL-2 value were reviewed in 2008 [L. 161] and 2010 [L. 163]. When compiling the catalogue and the accompanying database of seismological data, the scope of the retrieved information was adjusted to comply with the requirements specified in Paragraph 3.26 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]). Tab. 120 contains an overview of the required information with references to the sources available in the Temelín NPP seismological database.

Tab. 120 Structure of information in the Temelín NPP seismological database.

Information required pursuant to Para. 3.26 of IAEA SSG-9		Reference to Temelín NPP seismological database
(a)	Date, time and duration of event	Catalogue in: [L. 132] - Taken from regional catalogues; durations are available only for some events - in specialised publications
(b)	Localization of macroseismic epicentre of earthquake	Catalogue in: [L. 132] - Taken from regional catalogues
(c)	Estimated focal depth	Catalogue in: [L. 132] - Taken from regional catalogues
(d)	Estimated magnitude of event	Catalogue in: [L. 132] - Converted to moment magnitude (Mw) for all events according to the conversion formulas valid for individual regions - see basic material [L. 88]
(e)	Macroseismic intensity of event	Catalogue in: [L. 132] - Taken from regional catalogues
(f)	Isoseist maps	Available only for certain, namely strong events - see basic material [L. 170], and/or in specialised publications
(g)	Intensity in the locality and/or data on recorded effects on the soil and landscape	Information on remote earthquakes perceived in the near region of Temelín NPP - see basic material [L. 161]
(h)	Estimated uncertainty in catalogue parameter determination	See regional source catalogues
(i)	Determination of catalogue data quality based on the scope and number of estimated parameters	See catalogue homogeneity evaluation in [L. 132]
(j)	Information on felt foreshocks and aftershocks	See regional source catalogues and information in [L. 161]
(k)	Information on faults associated with individual shocks	See basic material [L. 160] and Fig. 29

[46] Institute of Rock Structure and Mechanics - Results of grant projects: IAA300120905 - "Dynamic of crustal fluids in the western part of the Bohemian Massif and its relation to stress changes" (2009-2012), GA205/06/1828 - "3D monitoring of micro-movements in the effective collision zone between African and Euro-Asian plates" (2006-2008) and GP205/08/P521 - "Manifestations of late Quaternary tectonics within the Sudetic Marginal Fault zone" (2008-2010).

The compiled catalogue of earthquakes (version 2011), i.e. the form used to determine the seismic hazards for the Temelín NPP site, contains events dating to the period from 1200 to 2010 and occurring in the area delimited by the coordinates: 46°-52°N; 10°-23°E. The catalogue includes Mw = 3.0 – 6.8 events and the total number of registered events amounts to 3,859 (see Fig. 26). Approximately from the year 1973, these events are instrumentally recorded.

The full version of the catalogue also contains 16 additional events recorded before the year 1200, the oldest dated to the year 456. Since their parameters are highly unreliable, they were not considered in the determination of SL-2.

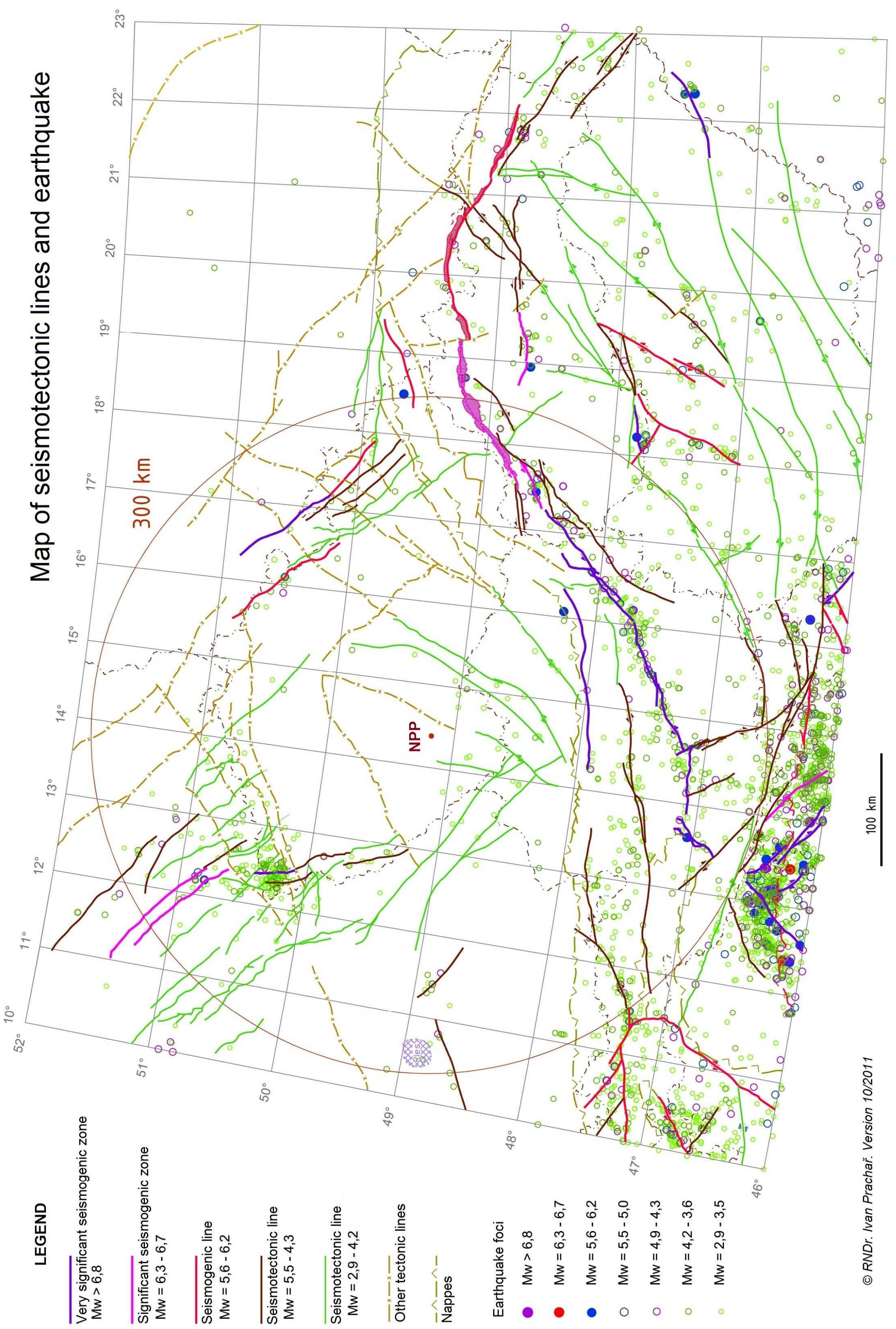

Fig. 29 Correlation between the course of known seismogenic lines and the occurrence of earthquake foci in the region of Temelín NPP according to the catalogue compiled by I. Prachař, version 2011 (in: [L. 132]).

Data on Instrumentally Recorded Earthquakes

The data along with information on historical earthquakes were either taken from regional catalogues or obtained from the seismological services of the countries falling within the region of Temelín NPP.

The scope of information required in relation to instrumentally recorded earthquakes is the same as for historical earthquakes (see Paragraph 3.26 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]). In addition, data recorded by broadband seismographs and accelerographs are required.

With respect to the near region of Temelín NPP, these data are available through the outputs of the monitoring of micro-earthquakes within the Local Temelín Network operated by the Institute of Physics of the Earth in Brno[47]. The occurrence of micro-earthquakes has been continuously monitored since 1991 [L. 92] in compliance with the recommendations stipulated in Section 2 (page 10) of IAEA TECDOC-343 [L. 16]. The map of the foci of historical earthquakes, instrumentally recorded earthquakes and micro-earthquakes shown in Fig. 41 was taken from [L. 161].

2.6.7.1.5 Requirements According to Paragraph 3.3 (NS-R-3)

This paragraph imposes an obligation to create a regional seismotectonic model (or alternative models) with the aid of available geological, geophysical, geotechnical and seismological data (see also Paragraphs 4.1 to 4.2, 4.6 to 4.7 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14]). Another significant input material is the earthquake catalogue, which should be evaluated during the initial phase of development of the seismotectonic model within the meaning of Paragraph 4.8 of the above mentioned IAEA Specific Safety Guide.

The subsequent step is to identify the sources of earthquakes and the delimitation of the source zones. To a greater or lesser extent, the region consists of two basic source area types: areas with seismogenic structures (i.e. faults causing earthquakes - causative faults) and areas with diffuse seismicity (see Paragraph 4.4 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14]). When delimiting focal areas, preference should be given to linear (quasilinear) areas embracing each significant seismogenic fault or fault system. However, where fault analysis suggests the presence of a number of parallel, accompanying faults or faults wingetpinnate to the main seismogenic structure, areas with concentrated seismicity should be delimited in addition to the quasilinear source areas.

A major step in the development of a seismotectonic model is the determination of the parameters of each source zone, namely the construction of the magnitude-frequency relationship curve (i.e. the Gutenberg-Richter relation with the formula: $log\ N_c = a - b.M_i$) - see Paragraphs 4.9 to 4.12 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14]. Various source zone parameters may be derived from the formula, such as the *a, b* constants and the *λ* parameter, which is defined as the mean value of the occurrence of an earthquake in a given period of time provided that Poisson distribution applies to the earthquake occurrence.[48] These parameters are subsequently used to determine (calculate) the maximum regional magnitude for

[47] See Seismological Information Display - Local Temelín Network http://www.ipe.muni.cz/newweb/cesky/index.php?main=siteastanice/siteastanice.php

[48] Poisson probability distribution applies to a random variable that expresses the number of occurrences of less probable, rare phenomena in a given time and/or volume interval.

each zone. As a rule, an exact determination of these parameters for areas with diffuse seismicity is not possible and expert estimates are implemented instead.

The final step is to determine the attenuation characteristics of the geological environment through which seismic waves propagate. If possible, preference should be given to an attenuation model that is based on an evaluation of the accelerograms recorded on the site of the nuclear installation.

With a view to the seismotectonic and seismological assessment of the region of Temelín NPP, we may state that the highest seismic load of the ETE3,4 site area is attributable to the energy released by remote earthquakes (more than 100 km from Temelín NPP), especially in the area of the Eastern Alps, which propagates into the Bohemian Massif with relatively low attenuation (cf. basic material [L. 162]). The influence of near earthquakes with magnitudes up to Mw = 4 (see basic material [L. 161]) is insignificant compared to the magnitudes of events occurring in the Eastern Alps area (cf. Fig. 21 in Section 2.6.2.1.2 showing the distribution of seismic load in the region of Temelín NPP in ground acceleration values).

An analysis of the seismicity of the region of Temelín NPP implies that:

- The most significant, seismically active structure in the region of Temelín NPP is the northwestern border of the ALCAPA block, represented by the Mur-Mürz-Leitha line and the Peripieniny lineament on the boundary of the Bohemian Massif and the Carpathians. Historically observed earthquakes on the Mur-Mürz-Leitha line reached a magnitude of up to Mw = 6.0 (Murau, 1201). The Friuli-Idrija area is also significant despite its distance of more than 300 km.
- Even though the subsurface contact of the Bohemian Massif and the Eastern Alps is not associated with such a distinctively delimited fault zone, a focus of a strong earthquake was localized here in more recent times (e.g. Riederberg, 1590).
- Most focal areas of Central Europe are characterised by focal depths ≤ 10 km, i.e. the earthquake hypocentres lie in the upper part of the Earth's crust where thickness rarely exceeds 28 - 40 km. The deepest hypocentres may be found in the area of Český Těšín and Bielsko-Biała (35 – 40 km).
- In the wider region, two types of earthquake sequences may be identified. The first are earthquake swarms represented by groups of weaker and stronger shocks where no shocks predominate in size. They are typical for the areas of Opava, Aš-Skalná-Kraslice, and Kunějov. The second is the mainshock-aftershock sequence where the main shock considerably exceeds the size of the aftershocks. Such sequences occur in the source areas of Hronov-Poříčí, Mur-Mürz-Leitha, etc. Only in rare cases, foreshocks are observed, i.e. weak shocks preceding the main shock.

Catalogue Evaluation

The compiled catalogue of earthquakes was compiled with the aid of regional catalogues and overviews of instrumentally recorded events from all countries falling within the region of Temelín NPP (Germany, Poland, Slovakia, Austria, Hungary, Romania, Croatia, Slovenia, Italy and Switzerland).

The conversion to uniform magnitude (i.e moment magnitude - Mw), as indicated in Paragraph 4.8 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14], was carried out during the compilation process. The reliability, homogeneity and the completeness of the catalogue were verified within the determination of the seismic hazard for the Temelín NPP site (see basic material [L. 132]). It was not possible to identify foreshocks and aftershocks for all catalogued events because the source catalogues do not contain such information, namely in relation to older earthquakes.

Delimitation of Source Zones and Determination of Source Zone Parameters

The source zones in the region of Temelín NPP were delimited by means of a method that strictly reflected the seismotectonic situation in the region (course of seismically active faults). Thus, largely quasilinear source zone and/or zones with concentrated seismicity (e.g. in areas with frequent earthquake swarms) were identified - see Fig. 31. A list of these source zones and their determined parameters are included in [L. 132].

Determination of Maximum Regional Magnitude

An advanced (K-S) computational model for deriving the maximum regional magnitude value was implemented according to [L. 118]. An expert estimate was used for zones for which parameters could not be determined.

Selection of the Attenuation Model

An analysis of the attenuation relationships utilised within previous calculations of seismic hazards for Temelín NPP is detailed in [L. 162] and [L. 131]. During the most recent revalidation of seismic hazards (see basic material [L. 173]), New Generation Attenuation (NGA) relationships were introduced. NGA relationships are based on a broader assessment of the effects, such as magnitude saturation, the relationship between attenuation and magnitude size, focal mechanism, the depth of the seismic energy source, hanging-wall geometry and the subsurface response at the investigated site. In this case, the Campbell and Bozorgnia [L. 66] attenuation model was used - see Fig. 30. The model was derived from an extensive set of world earthquakes, for which instrumentally recorded accelerograms were available. Its applicability was verified by a comparison of the attenuation curves with available accelerometer data recorded by stations that are a part of the local seismic network in Temelín and of the national network of seismic stations operated by the Institute of Geophysics of the Academy of Sciences of the Czech Republic. A local attenuation model cannot be developed due to the lack of accelerometer data particularly of strong earthqaukes, recorded by the Temelín seismic network.

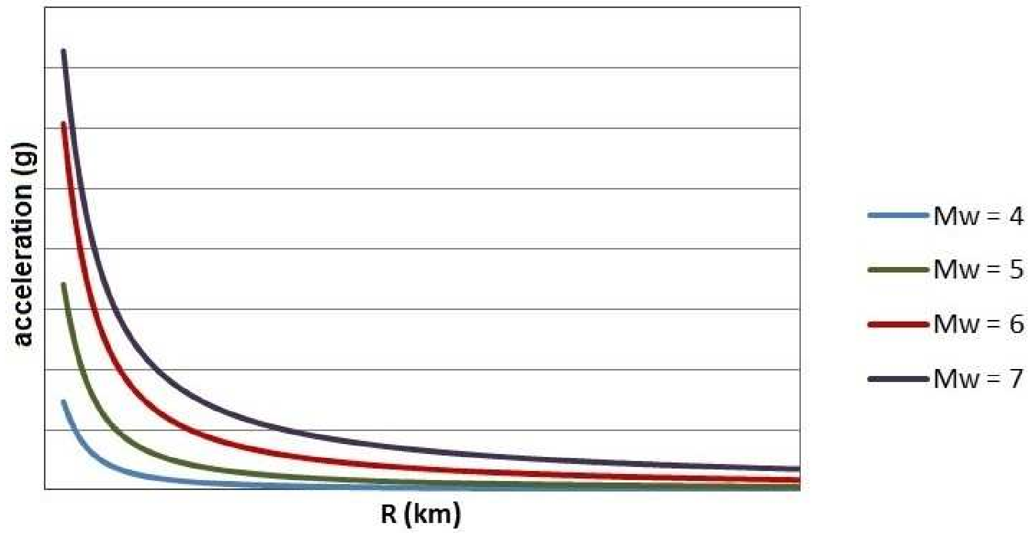

Fig. 30 Seismic energy attenuation curve based on the Campbell and Bozorgnia model [L. 66] for magnitudes Mw = 4, 5, 6 and 7. Mean value - does not include bedrock response. The relationship is valid for R distances ranging from 0 to 200 km.

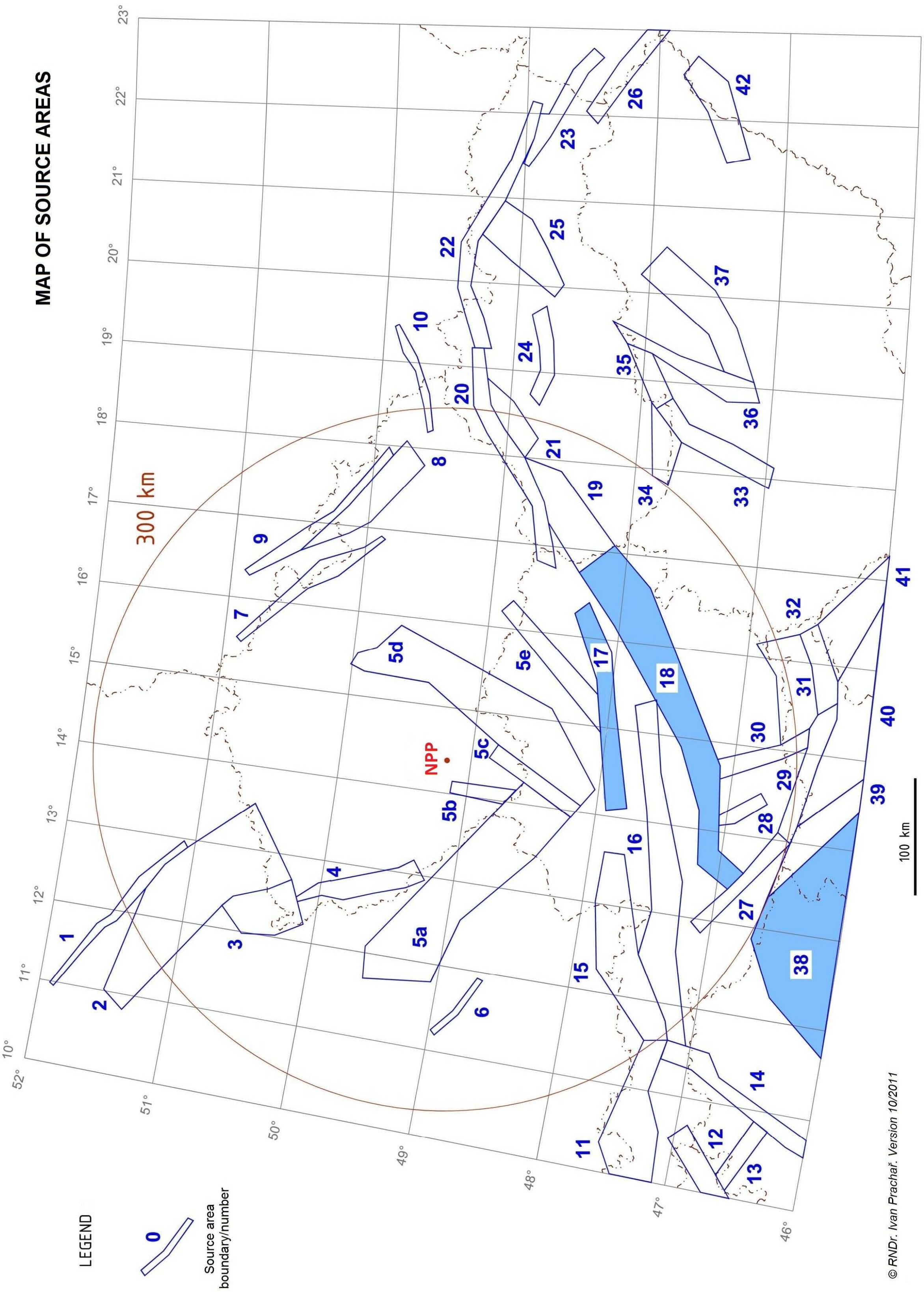

Fig. 31 Source zones of earthquakes in the region of Temelín NPP. The zone names corresponding to the relevant numbers are specified in [L. 132]. The light-blue segments represent zones that are the most significant source of seismic load in the Temelín NPP site (17 - Neulengbach lineament; 18 - Mur-Mürz-Leitha fault system; 38 - Friuli-Tolmin-Idrija area.

2.6.7.1.6 Requirement According to Paragraph 3.4 (NS-R-3)

As regards the identification of seismic hazards for a nuclear installation (i.e. determination of SL-2), two main approaches are applied: the deterministic method, which defines marginal conditions (Seismic Margin Assessment - SMA), and the probabilistic method (Seismic Probabilistic Safety Assessment - SPSA) - see also Paragraph 5.1 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14].

As stipulated in Paragraph 5.2 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14], an integral part of seismic hazard evaluation should be an assessment of uncertainties. During a seismic hazard analysis, two types of uncertainties should be taken into account – i.e. aleatory variability and epistemic uncertainty.

Aleatory variability is defined as the natural randomness in a process. It is typically characterised as the standard deviation of a regression coefficient in analysis computational models.

Epistemic uncertainty is defined as the scientific uncertainty in the model of a process, which is given by limitations both in data and in our knowledge. Epistemic uncertainty may be described by alternative models. Paragraph 6.4 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14] recommends the development of a logic tree as an acceptable method for dealing with epistemic uncertainties within the probabilistic analysis. The logic tree may be evaluated either by means of a complete enumeration of the logic tree branches or by the Monte Carlo simulation. A seismic hazard curve should be used to display epistemic uncertainty – namely 16%, 50% and 84% quantiles.

Determination of SL-2

In the IAEA Safety Guides, SL-2 is defined as "*the most stringent safety requirement*" (see Paragraph 9.1 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]). In practice, this level is associated with the size of ground motion that may occur with small probability. Some countries accept the probability that this value may be exceeded in the range from 1×10^{-3} to 1×10^{-4} (mean value) or from 1×10^{-4} to 1×10^{-5} (median) in one year - see Paragraph 2.3 the IAEA Safety Guide No. NS-G-1.6 - [L. 19]).

In compliance with the recommendations specified in Sections 6 and 7 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14], a revalidation of the seismic hazard was completed in 2011 (see basic material [L. 132]). The probabilistic method (SPSA) was used as the fundamental approach, whereas the steps defined in Paragraph 6.3 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14] were executed as follows:

- Shocks were categorized according to their source zone in a manner strictly reflecting the seismotectonic situation of the region (course of seismogenic faults). Zone boundary uncertainties were resolved by comparing the zone boundaries with geological findings on each fault system, as well as by a comparison of the delimited zones with the distribution of the foci of historical earthquakes;
- The maximum regional magnitude was derived with the use of a "more advanced" (K-S) procedure according to [L. 118], including the determination of the variation value;
- Aleatory variability was evaluated by computational models in each stage and solved by including the standard deviation σ instead of increasing the input parameters by the safety margin;

- Knowledge-based (epistemic) uncertainties were expressed through seismic hazard curves – for 16%, 50% and 84% quantiles;
- SL-2 was determined up to the annual frequency of 10E-5;
- The resultant SL-2 calculated by implementing the probabilistic approach was verified by a neo-deterministic approach using synthetic seismograms that were computed for certain source zones by means of the "reflectivity" method. The relationship between seismograms and focal depth, focal mechanisms and the model of the environment between the source and the Temelín NPP site were also investigated (see basic material [L. 132]);
- The results were checked by simplified calculations with the aid of the parametric-historical method (see basic material [L. 132]).

The resultant seismic hazard curves are shown in Fig. 32.

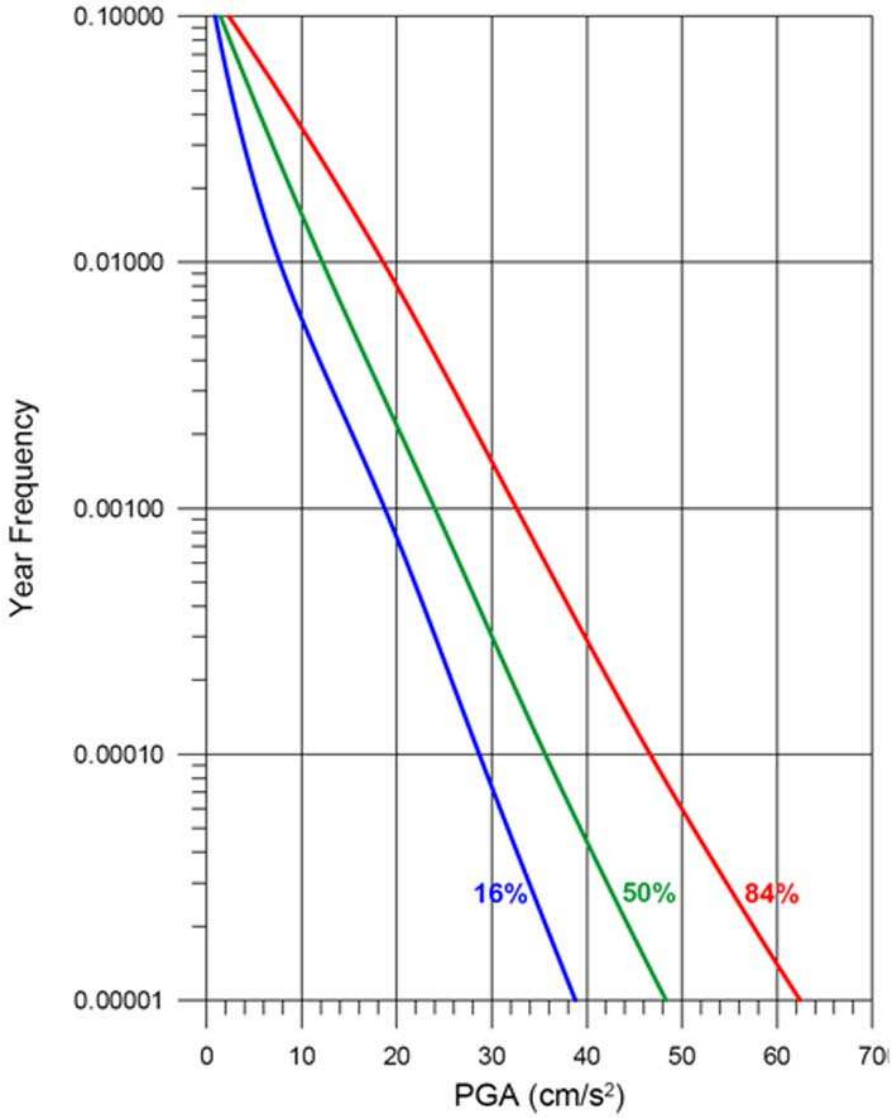

Fig. 32 Seismic hazard probability curves for the Temelín NPP construction site.

As shown in Fig. 32 above, the most probable scenario (50% curve) is that, in a period of 10,000 years, the ETE3,4 site area will not be exposed to seismic ground motion with accelerations of more than 36 cm/s^2. The figure also suggests an 84% probability that the value of 47 cm/s^2 will not be exceeded over the period of 10,000 years. For the period of 100,000 years (frequency 10E-5), the following values apply: 48 cm/s^2, respectively 62 cm/s^2.

It is possible to state that previously performed calculations (deterministic using macroseismic intensity values - see Section 2.6.7.1.2, as well as probabilistic (indicated in the latest version of the POSR of Temelín NPP [L. 18] and taken from [L. 173])) were correct and they did not deflect from the spread of realistic evaluation results.

With respect to evaluations mentioned in [L. 173] and [L. 132]), the result difference is very small and for the 50% probability and the annual frequency of 10E-4 it is essentially the same (37 cm/s^2).

Determination of SL-1

In the IAEA Safety Guides, SL-1 is defined as a "*less severe, more probable earthquake level*" (see Paragraph 9.1 of the IAEA Specific Safety Guide No. SSG-9 - [L. 14]). In practice, this level is associated with ground motion of a size to which the nuclear power plant staff is likely to be exposed with high probability. In some countries, it corresponds to the level with a probability of being exceeded in the range of 100 years (frequency 10E-2) - see Paragraph 2.3 of the IAEA Safety Guide No. NS-G-1.6 [L. 19]).

Considering the most probable scenario (50% curve), SL-1 for the site area of Temelín NPP corresponds to 14 cm/s^2. For comparison purposes, Tab. 121 presents the ground accelerations induced by known, very strong earthquakes in the region of Temelín NPP (Campbell and Bozorgnia attenuation model [L. 66], 50% probability).

Tab. 121 Ground acceleration on the Temelín NPP site area induced by known strong earthquakes.

YYYY	MM	DD	HH	Mi	°N	°E	h (km)	Mw	I_0	Focus	$R_{(Temelín\ NPP)}$ (km)	PGAhor (cm/s^2)
1348	1	25	16	0	46.40	13.40	8	6.8	10.0	CHIUSAFORTE	317	10
1511	3	26	14	0	46.10	14.00	15	6.8	10.0	IDRIJA-CEKRNO	343	9
1590	9	15	23	50	48.26	16.07	6	5.8	9.0	RIEDERBERG	161	9
1511	3	26	20	0	46.20	13.43	20	6.5	10.0	FRIULI	338	7
1201	5	4	14	0	47.05	13.62	8	6.1	9.0	KATSCHBERG	243	7
1976	5	6	20	0	46.28	13.10	17	6.4	9.5	FRIULI-GEMONA	336	7
1873	6	29	3	58	46.15	12.38	-	6.3	9.5	BELLUNESE	367	6

Response Spectra

The seismic assessment of the nuclear installation construction site also includes an analysis of the response of the foundation soil to seismic load. The result is a curve known as a response spectrum.

The response spectrum was developed for the so-called control earthquake, i.e. the strongest earthquake recorded in the region of Temelín NPP - a paleoseismologically indicated shock near the Siehdichfür farm (Austria,zone No. 18) with Mw = 7.0. It was plotted with the aid of the Campbell and Bozorgnia attenuation model coefficients [L. 66] for the period from 0.01 to 10 s. The resultant response spectra curves for the mean value (50%) and the mean value + σ (84%) are provided in Fig. 33.

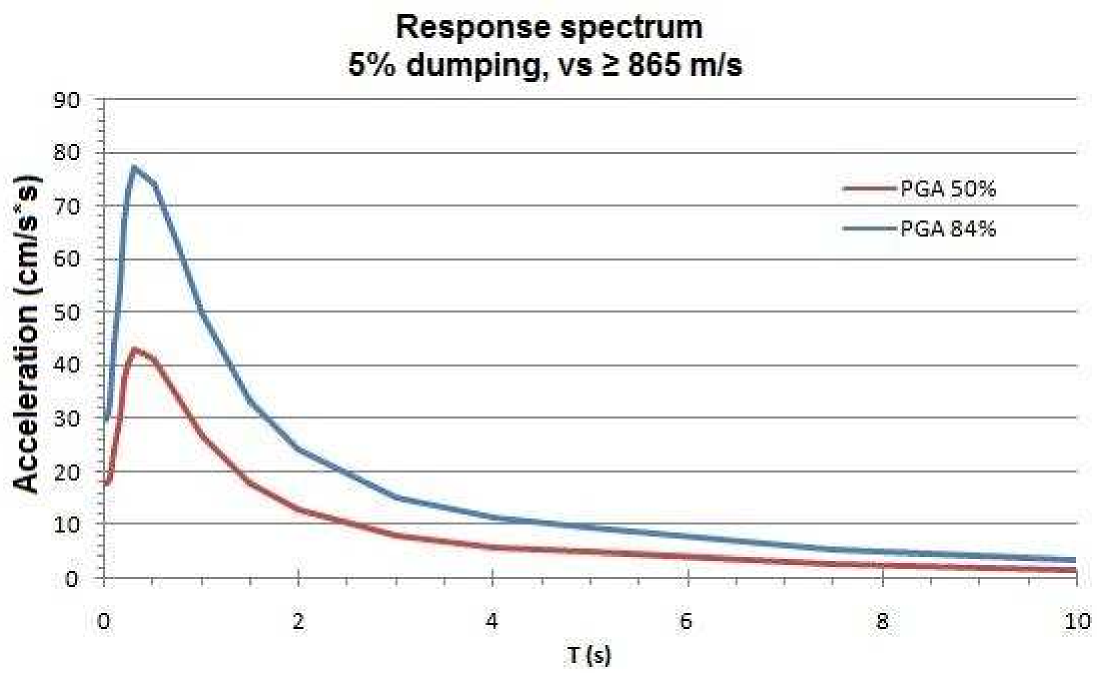

Fig. 33 Response spectra for the construction site of Temelín NPP plotted based on attenuation model coefficients according to [L. 66] (taken from [L. 132]).

When formulating the seismic design of a structure, it is more advantageous to choose broadband response spectra derived from standard curves (e.g. NUREG/CR-0098 - [L. 141]). This spectrum, converted to PGA 0.1 g, was used in the seismic design of Temelín NPP1,2 (see basic material [L. 18]). The selection of the standard spectrum is given by the location of Temelín NPP in an area with low to medium seismicity even if the response curves should be broadband and site specific in accordance with Paragraph 4.2 of the IAEA Safety Reports Series No. 28 [L. 33]).

Ground Motion Duration

The duration of the main ground motion phase was estimated based on an evaluation of the accelerograms of earthquakes recorded in the region of Temelín NPP. Both records from the databank of strong earthquakes (see lit. [L. 49]) and from the accelerometer at the Struha station (Local Temelín Network - see footnote 47) were used. The relevant accelerogram is shown in Fig. 34.

The evaluated accelerograms imply that the duration of the maximum phase of ground motion should reach up to 10 s (4 - 10 s).

2.6.7.1.7 Effects of Induced and Technical Seismicity

Induced seismicity is mentioned on page 9 of IAEA TECDOC-343 [L. 16]) as one of the recommended micro-earthquake monitoring outputs. The necessity to evaluate the induced seismicity hazards also arises from Paragraph 8.7 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]), namely in connection with the recommended monitoring of suspicious, potentially capable faults. According to the IAEA Safety Specific Guide, also faults that, despite not demonstrating any geological manifestations, may be reactivated by reservoir loading, injection of water in the fault structure, withdrawal of water or other fluids from the structure, etc. should be considered suspicious. In these cases, the occurrence of micro-earthquakes in the vicinity of such structures may serve as another major indicator.

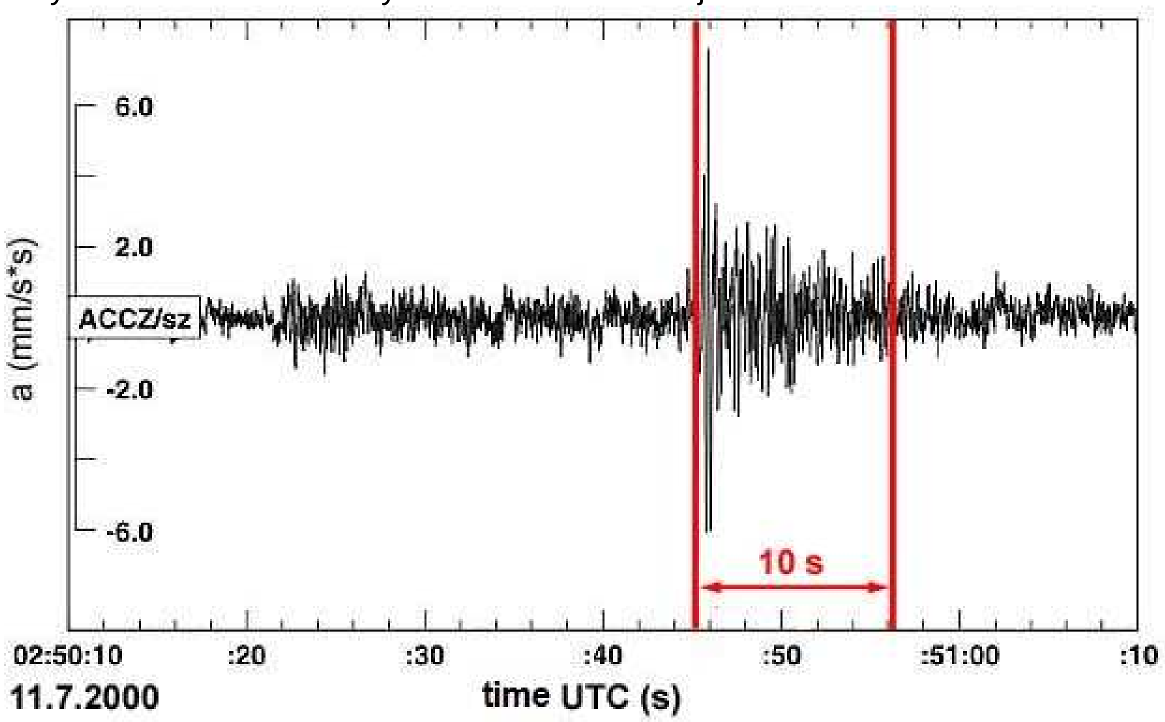

Fig. 34 An accelerogram of an earthquake that occurred on 11 July 2000 at 4:50 a.m. Central European Summer Time (CEST) near the municipality of Ebreichsdorf, approx. 25 km south of Vienna. The magnitude of the event reached ML = 4.8. The recording was made by the Struha station (Local Temelín Network). Taken from http://www.ipe.muni.cz/newweb/cesky/index.php?main=siteastanice/siteastanice.php

As mentioned above, induced seismicity is generated in the vicinity of large artificial reservoirs, especially in case of water level changes, in the surroundings of oil and gas fields, in the vicinity of underground gas tanks or in the surroundings of areas where fluid is injected into geological structures. It is also caused by the collapse of underground cavities (mines, caves), mountain bumps (pillar bursts). Artificially induced seismicity is generated by blasting in quarries and on construction sites.

Induced seismicity is recorded by the local seismic network in Temelín, particularly in the area of the Háje underground gas storage, which is located approximately 60 km from Temelín NPP. The magnitude of the strongest shocks reached $M_L \approx 2$. In total 56 events were recorded throughout the period of operation of the network (1991 - 2011).

A plentiful group of epicentres (66 since 1991) was recorded in the area of the Orlík Reservoir (Klučenice - Milešov - Kozárovice). The focal distance from Temelín NPP is approximately 45 km. The strongest recorded magnitude was $M_L = 2.3$ (event occurring on 13 January 2007). The origin of micro-earthquakes in the surroundings of the Orlík Reservoir has not been fully clarified for the time being. One of the possible explanations speaks of water penetration into the surrounding rock mass or into existing fault or disturbance zones (see lit. [L. 91]).

As regards the manifestations of industrial quarry blasting on the Temelín NPP site area, the strongest effects originate from the quarry in Slavětice, which is located approximately 5.8 km from Temelín NPP. The magnitude of the induced shocks ranges around $M_L \approx 1$.

2.6.7.1.8 Summarized Evaluation

Maximum Calculated Earthquake (MCE) determined through several methods, by various investigators and with the use of different seismotectonic models reached the maximum value of 6.5° MSK-64 [L. 18]. Likewise, the results of the newly performed revalidation of the seismic hazards of the Temelín NPP site [L. 132], which determined SL-2 in the form of Peak Ground Acceleration, do not imply any exceedance of the limit values defined by Decree No. 215/1997 Coll. [L. 1].

Thus, it is possible to say that the land of the proposed siting of ETE3,4 is not exposed to seismic load expressed by MCE, which would exceed the limit values stipulated either by the exclusion criterion under Section 4(e) or the conditional criterion under Section 5(c) of Decree No. 215/1997 Coll. [L. 1].

Furthermore, the content of Sections 2.6.7.1.3 to 2.6.7.1.6 indicates that the scope and detail of the information gathered in the geological and seismological database of the region of Temelín NPP comply with the requirements set out in Paragraphs 3.1 to 3.3 of the IAEA Safety Standard No. NS-R-3 [L. 6]. The determination of the seismic hazard (see basic material [L. 132]) constitutes the fulfilment of the requirement stipulated in Paragraph 3.4 of the IAEA Safety Standard No. NS-R-3 [L. 6].

In compliance with the recommendations specified in IAEA TECDOC-343 [L. 16]), micro-earthquakes are monitored in the surroundings of ETE3,4. The recorded data are evaluated by the Institute of Physics of the Earth, Masaryk University Brno, and documented in annual reports concerning the "Detailed Seismic Zonation of the Surroundings of Temelín NPP" (see also [L. 92]).

It is possible to say that sources of induced and technical seismicity are monitored in the surroundings of ETE3,4. The monitoring results suggest that the ETE3,4 construction site is not jeopardized by these types of seismicity.

2.6.7.2 CRITERION AND REQUIREMENTS ACCORDING TO SECTION 7.2

2.6.7.2.1 Specification of Hazards Grouped in Section 7.2

Section 7.2 in Tab. 115 groups together the geological phenomena that represent a jeopardy of fault displacement on the construction site of a nuclear installation or in its close surroundings. When assessing this criterion, primary emphasis is on the ability to identify a fault potentially capable of displacement (capable of generating an earthquake), i.e. a capable fault.

In general, a fault is a disjunctive fracture of a geological body with a macroscopically visible and significant displacement relative to the size of the disrupted body [L. 152]). From the point of view of nuclear safety, the subject of interest are only some faults that are causally linked to fault displacement and/or to earthquake occurrence. A fault capable of displacement (i.e. capable fault) is defined as a fault that shows significant potential for displacement at or near the ground surface. A seismogenic structure is such tectonic structure (fault), which is associated with the foci of earthquakes, past surface faulting or signs of paleoseismicity and where macro shocks will probably occur also in the period under review.

When assessing this criterion, it is necessary to keep in mind that it concerns displacements on faults induced, whether directly or indirectly, by earthquakes. A displacement may be directly connected to the fault where an earthquake occurred or it may appear in consequence of an earthquake on a secondary fault. Therefore, the entire zone of an active fault should be evaluated.

The criterion defined in Section 4(f) of Decree No. 215/1997 Coll. [L. 1] excludes the siting of a nuclear installation on a site where there is a jeopardy of a change in terrain morphology, surface deformations in the area and of structural damage caused by the displacement of blocks of the Earth's crust on existing faults or on new faults formed as a result of an earthquake.

The IAEA Safety Guides give great attention to the issue of displacements on faults on nuclear installation sites. The jeopardy of potential fault displacement on the construction site is a definitive exclusion criterion. Paragraph 3.7 of the IAEA Safety Requirements IAEA NS-R-3 [L. 6] provides: "Where reliable evidence shows the existence of a capable fault that has the potential to affect the safety of the nuclear installation, an alternative site shall be considered."

A key recommendation related to fault activity is included in Paragraph 8.6 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]: "When faulting is known or suspected to be present, site vicinity scale investigations should be made that include very detailed geological and geomorphological mapping, topographical analyses, geophysical surveys (including geodesy, if necessary), trenching, boreholes, age dating of sediments or faulted rock, local seismological investigations and any other appropriate techniques to ascertain the amount and age of previous displacements."

The determination of whether "faulting is known or suspected" on the site is the most difficult stage of the evaluation process. The resolution of this question is closely linked to the specific conditions of the geological units located in the areas under review. It was thus necessary to develop "area-specific" systems of auxiliary criteria that would allow to define the above mentioned "suspicion".

The "suspicion" (in conditions of the Bohemian Massif) was defined with the aid of system of own auxiliary criteria, which includes:

- Occurrence of linear topographic or structural relief elements (fault slopes, straight running slopes, lineaments);
- Occurrence of distinct lithological boundaries, particularly with the presence of sedimentary units of the platform cover
- Occurrence of rocks - the products of mechanical rock deformation along tectonic lines or the occurrence of clay and other minerals formed in near-surface conditions;;
- Occurrence of micro-earthquakes or local historical earthquakes, especially in geographic coincidence with the specified characteristics.

The evaluation of items 1 to 4 may also include the modelling of the relative reactivation potential of the faults in the area under review.

The evaluation according to this criterion should start with an analysis of the wider area (at minimum covering the site of the nuclear installation) within the meaning of Paragraphs 3.11 to 3.18 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]). The above named paragraphs provide recommendations for the evaluation of the tectonic history and the present tectonic regime of the near region of a nuclear installation. In case of the near region of ETE3,4, with reference to Paragraph 3.12 of the IAEA

Specific Safety Guide, the evaluation should cover the Pliocene-Recent period. The evaluation should focus:

- On determining the past and present tectonic regime;
- On assessing past movements on faults, their nature and frequency, on dating the most recent movements on faults, as well as on documenting records of prehistoric earthquakes ascertained through paleoseismological research;
- On the occurrence of local historical, instrumentally recorded earthquakes and micro-earthquakes and on determining their relationship to known fault structures.

2.6.7.2.2 Recommendations According to Paragraphs 3.11 and 3.12 of the IAEA Specific Safety Guide No. SSG-9

Characteristics of the Past and Present Tectonic Regime

The knowledge of the development of the stress condition in the wider area (region) of a nuclear installation is an essential precondition for the understanding of the past and present tectonic regimes. The characteristics of Cenozoic[49] paleostress fields and their succession in the Bohemian Massif were investigated mainly by paleostress analysis of Cenozoic rock brittle failure. The resultant picture was complemented with the results obtained by applying the same analysis to rocks of Variscan age where a part of the brittle failure is considered to be of post-Variscan (to Cenozoic) age.

The Cenozoic paleostress fields found in the Bohemian Massif may be divided into four groups (see basic material [L. 188] and [L. 168]):

- Compression fields of Upper Cretaceous to Eocene age, which are undoubtedly reflections of north-vergent collisions in the Alpine area. Fault activation in the Bohemian Massif is largely of an overthrust or strike-slip character, with documented formations of extensive fold-fault structures in the period in the northeastern and eastern part of the massif.
- Predominantly extension stress fields of Eocene to Middle Miocene age that are genetically linked with the east-vergent lateral extrusion of the units of the Eastern Alps and with the activation of the European Cenozoic Rift System in the updoming area in the Alpine foreland. A number of extension stress fields associated with the development of the Ohře Rift are documented in the northern part of the Bohemian Massif. Stress directions changed in each development phase, as well as the individual parts of large structures. Along the eastern edge of the massif, normal-slip faults and sediment-filled depressions were formed. They are considered to be the consequence of NE-SE extension occurring in this period.
- Post-Lower Miocene compressions aligned in the ENE-WSW and NW-SE directions are demonstrably dated to the period of the Middle to the Upper Miocene. Their formation was accompanied by transverse disruption of the

[49] The Cenozoic - A geological era embracing the Tertiary and the Quaternary. The lower boundary of the Cenozoic is dated to 65 - 67 million years (see Encyklopedický slovník geologických věd [Encyclopaedic Dictionary of Geological Sciences], 1983).

Ohře Rift structure and activation of the reverse faults in the northeastern and eastern parts of the Bohemian Massif.

- From the Pliocene to the Quaternary, the Bohemian Massif was probably exposed to alternating and relatively short-term stress of variable character and orientation. Standing out prominently among them is the Pliocene extension of the NE-SW direction, which is namely apparent from the character of the deposition sediment centres in the Cheb Basin and the fault failure in the area of the Upper Moravian Vale.

The orientation of recent stress fields was ascertained in boreholes with the use of breakouts[50], overcoring[51] and hydrofracturing[52], as well as through an analysis of the focal mechanisms of weak natural(tectonic) earthquakes. The results of the study of brittle failure kinematics measured by a dilatometer may be put to use when considering the orientation of recent stress in the northeastern part of the Bohemian Massif. In a majority of the cases, NW-SE compression was ascertained and this corresponds to the prevailing trend in the Central European Alpine foreland.

An overview of the Cenozoic paleostress and recent stress characteristics identified in the Bohemian Massif is provided in [L. 188] (see Fig. 2.4-2).

The World Stress Map (WSM), which was plotted based on tectonic data within the International Lithosphere Programme (ILP) in 1995, may serve as an additional source for the delimitation of the general direction of the stress field. The map along with source data are available in [L. 94], [L. 95].

Another significant characteristic necessary for the development of the seismotectonic model and for the profiling of the tectonic regime is the size of the mutual horizontal and vertical movement of the individual domains in the subject region. [L. 86] implies that the East-European Precambrian platform, the German-Polish depression and the Bohemian Massif are stable and rigid structural domains in the region characterised by mutual horizontal intrablock movements with a speed of less than 1 mm per year. For further details and descriptions of recent movements in other areas of the region of Temelín NPP see [L. 160].

In terms of recent vertical movements, uplift tendencies in the Bohemian Massif are particularly apparent in its northern and northwestern parts. A pronounced uplift appears in the neovulcanite area of the Doupov Mountains [Doupovské hory] and in the area of the Lužice Pluton. Uplift tendencies may also be found in the area of the Šumava Mountains and the Bohemian-Moravian Highlands [Českomoravská vrchovina]. The central part of the Bohemian Massif, the area of southern Bohemia and Moravia, and the main part of the Carpathian Foredeep manifest subsidence tendencies (see Section 9.1 of [L. 64]). An assessment of recent vertical movements in the near region of Temelín NPP and its wider surroundings is specified in [L. 187] (see Part A, Section 6.2). See also Section 2.6.7.3.2.

[50] Borehole breakouts - This method is based on monitoring the extension of the borehole cross-section in response to preferential rock massif disruption.

[51] This method involves measurements of the rock massif stress tensor - CCBO (Compact Conical-ended Borehole Overcoring)-

[52] This method is used to measure massif stress (horizontal stress) by means of borehole wall hydrofracturing (Hydrofracturing Stress Measurements).

Evaluation of Past Movements on Faults

Expert literature and fieldwork in the near region of Temelín NPP have identified the following faults of regional significance (see Tab. 122), which considerably influenced the geological structure of the near region during the development of the Bohemian Massif.

Tab. 122 Overview of faults of regional significance in the near of Temelín NPP with specifications of the directional course and minimum distance from Temelín NPP.

Fault	Directional course of fault	Minimum distance from ETE3,4
Vodňany Mylonite Zone (6)	NE-SW	3 km
Blata Fault (7)	NW-SE	8 km
Vlhavy Fault (assumed) (8)	N-S	9 km
Líšnice Fault (5)	NNE-SSW	9 km
Munice Fault (4)	NNE-SSW	10 km
Blanice valley fault (9)	N-S	10 km
Hluboká Fault (3)	NW-SE	13 km
Drahotěšice Fault (1)	NNE-SSW	15 km
Rudolfov Fault (outside the near region) (2)*	NNE-SSW	25 - 33 km
Dubné Fault (outside the near region) (10)*	NW-SE	20 - 33 km

*) Without description in the following text.

The faults were subjected to paleoseismological research, which was carried out at suitable outcrops and in exploratory trenches. The most extensive investigations were performed on the Hluboká Fault (see [L. 188]).

Vodňany Mylonite Zone

The Vodňany Mylonite Zone is aligned in the NE-SW direction in the near region of Temelín NPP and it represents a facial boundary between the Podolí Complex and the Variegated Group of the Moldanubicum in the northwest and the zone of re-foliated gneisses of the Monotonous Group of the Moldanubicum in the southeast (see Fig. 35). It runs along the southern edge of the ridge of the Mehelník Highlands [Mehelnická vrchovina] and it is morphologically pronounced namely in the section between the villages of Fanfíry and Kaliště. In the section between Přehájek and Neznašov, the boundary is covered by Miocene sediments of the small basin of Bohunice and as it extends NE, it predisposes the valley of the Lužnice River. A detailed description may be found in [L. 127].

The Vodnaňy Mylonite Zone is founded on an older, deep zone of the character of an extensive ductile thrust. It was later modified by younger brittle tectonics with an almost 30° deflection from the older metamorphic structure accompanied by the formation of a mylonite zone aligned in the NE-SW direction. This mylonite zone is approximately 1 km wide and declines 30° to 40° to the northwest.

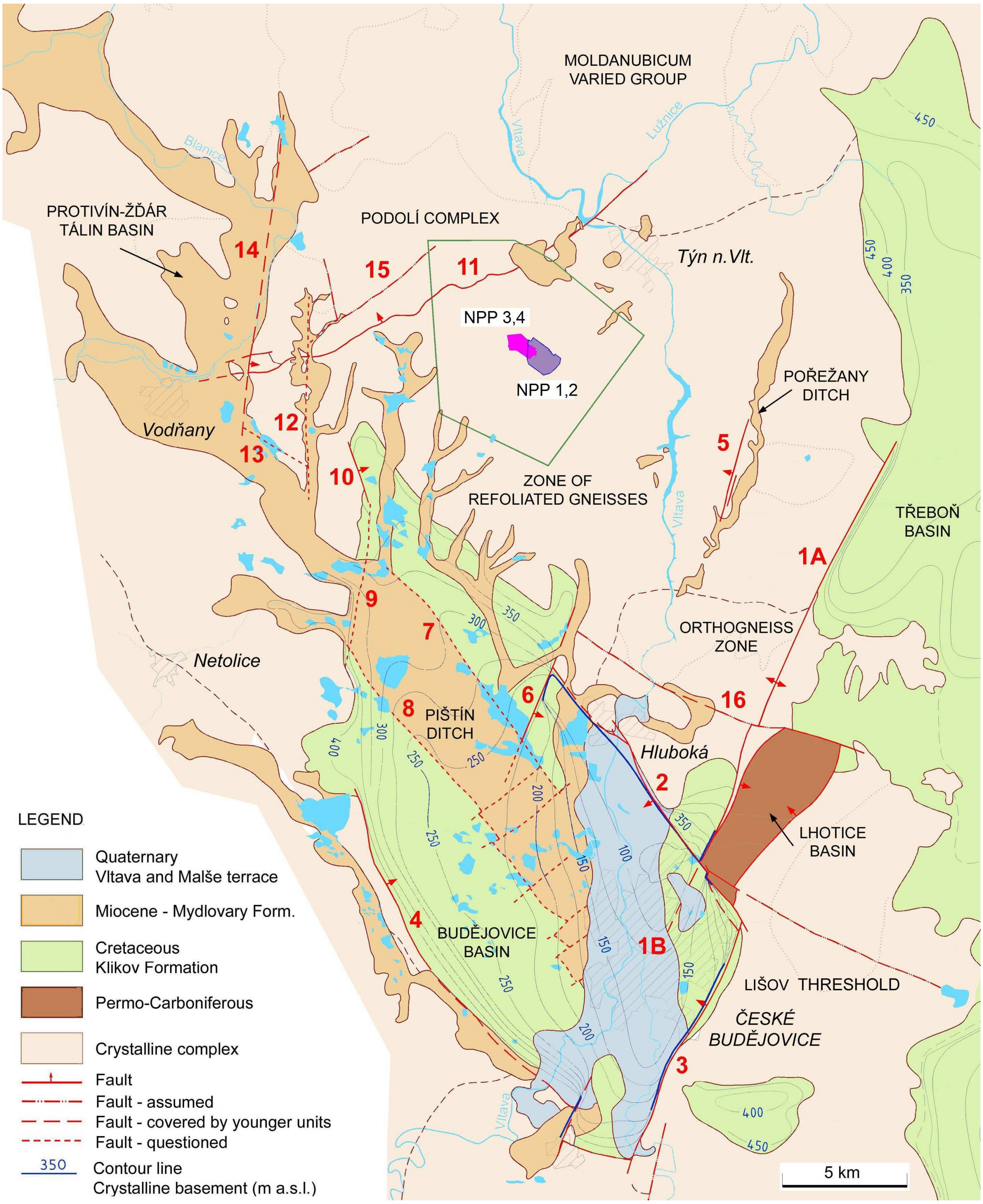

Fig. 35 Structural and tectonic map of the near region of Temelín NPP showing the main crystalline platform cover units.

Fault identification: 1 - Drahotěšice Fault, 2 - Rudolfov Fault, 3 - Hluboká Fault, 4 - Munice Fault, 5 - Líšnice Fault, 6 - Vodňany Mylonite Zone, 7 - Blata Fault, 8 - ***Vlhavy Fault****, 9 - Blanice valley fault. Identification of investigated sites - documentation points: I - Fanfíry trench, II - Outcrops near Čavyně, III - Záblatí trench, IV - Líšnice trench, V - Poněšice trench, VI - Outcrop in the valley of the Budáček Creck, VII - Deep boreholes near Munice, VIII - Trench and ERT profiles near Čičenice and Strpí, IX - Úsilné R1 and R2 trenches, - X - Munice trench, XI - Water pipeline ditch near Úsilné.*

Tectonic deformation of rocks is manifested by intense protoclasis and mylonitisation of leucocratic migmatites near Záboří and in the surroundings of Těšínov, by mylonitisation of quartzites near Lhota pod Horami and Fanfíry, by cataclasis of two-mica tourmaline orthogneiss near Čavyně, or by re-foliation of paragneisses in the northeast surroundings of Týn nad Vltavou (see lit. [L. 195]). The character of the zone is best visible on the outcrops located in the abandoned quarries near the municipality of Čavyně.

Even though the Vodňany Mylonite Zone is undoubtedly of old age, it was reactivated a number of times throughout its geological history. The most recent reactivation is probably of late Variscan dating (cf. lit. [L. 119]). Evidence of possible Pliocene and Pleistocene movement activity of the Vodňany Mylonite Zone was investigated at two localities. The first was the base of a morphological slope near the municipality of Fanfíry (outside the site vicinity area) where an exploratory trench was excavated (DB-I in Fig. 35, see lit. [L. 187]). A more pronounced morphological manifestation in the surroundings of the municipality of Fanfíry (see geomorphological map – basic material [L. 187] may arise from the coincidence of the margin of the Vodňany Mylonite Zone with the presumed edge of the Miocene sedimentation area.

The second locality was the approximately 300 m long rock escarp uncovered in the quarries near the municipality of Čavyně (DB-II in Fig. 35, see lit. [L. 168]). At both localities, fieldwork revealed cataclased and mylonitised rocks belonging to the Vodňany Mylonite Zone with plentiful slickensiding, striation on the planes, as well as signs of sericitisation and tourmalinization (see Fig. 36). The age of the tectonic rock deformation observed in the outcrops of the Vodňany Mylonite Zone (see e.g. Fig. 36) is probably Variscan. The outcrops revealed none of the characteristics of tectonic distortion that are typical of "young" tectonic movement in near-surface conditions.

Fig. 36 View of the wall of the abandoned quarry near Čavyně where a part of the Vodňany Mylonite Zone was found. The picture clearly shows a steeply positioned joint of the N-S direction. The joint uncovers a younger, brittle shear zone, which contains a "foliation fish". The yellow arrows show the mutual movement of the upper and bottom blocks. Photo by P. Rajlich, 2011. See also [L. 168].

Blata Fault

The Blata Fault is the name of a fault uncovered in the course of paleoseismological investigations carried out on the faults aligned in the N-S direction in the near region of Temelín NPP (see lit. [L. 167]). The fault was discovered while the researchers were verifying the course of the Vlhavy Fault in the section between Sedlec to the south and Záblatí to the north, particularly at the southern edge of the municipality of Záblatí.

With a 332/61 direction and inclination, the fault disrupts the Cretaceous sediments of the České Budějovice Basin. The fault plane is accompanied by an ferruginous sandstone tongue (see DB-III in Fig. 35 and Fig. 37). The character of the Cretaceous sediments found in the ZA-1 drill (see lit. [L. 167]) in the subsided northeastern block (fine gravel, clayey fine gravel, coarse-grained, quartz-feldspar sands) indicates that the function of the fault was most likely synsedimentary. The throw height is about 30 m.

The course of the fault to the northwest and to the southeast is not documented sufficiently. In the northwest direction, the fault probably runs through the residential area of the municipality of Záblatí and restricts the northwestern headland of the České Budějovice Basin. The alignment of the southeastern course of the fault is more difficult. No morphological manifestations were detected either on the surface or in the drills drilled to the depth of the crystalline basement and no pronounced throw is visible. We therefore presume that the course of the Blata Fault to the southeast, if it continues in that direction at all, is associated with a substantial lowering of the fault throw.

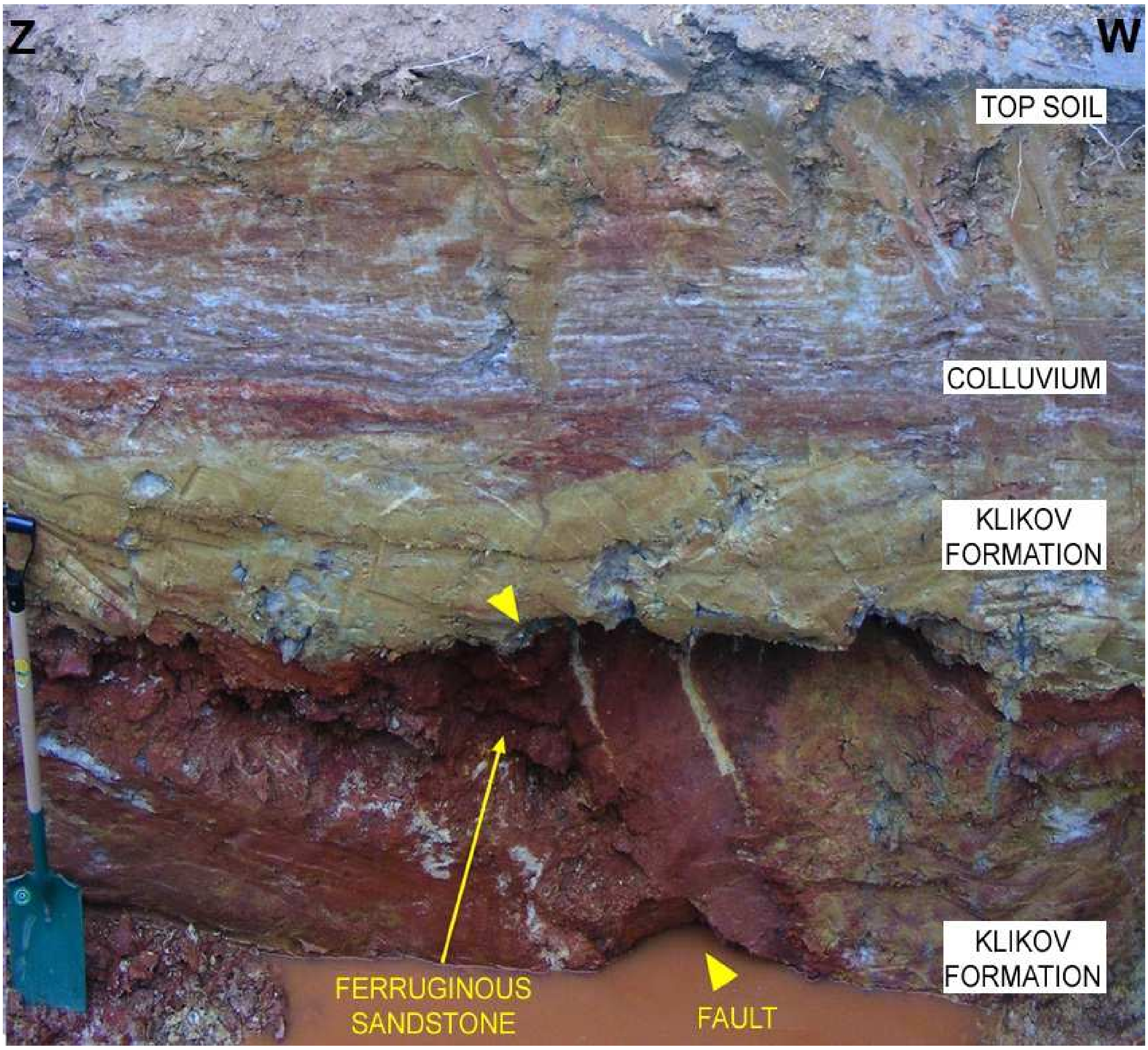

Fig. 37 View of the northern wall of the trench near Záblatí showing the fault plane of the Blata Fault and its position relative to the sediments of the Klikov Formation, the colluvium and the Holocene soil cover. Photo by I. Prachař, 2011.

Líšnice Fault

The Líšnice Fault was first described in [L. 50]. The fault is aligned in the NNE-SSW direction and its course may be traced from Bečice in the north to Ponešice in the south. It consists of a number of parallel dislocations with mylonites, the thickness of which ranges from a few metres to dozens of metres (see lit. [L. 134]).

The course may be divided into three sections: the first north section running between Bečice and Pořežany, the second middle section running between Pořežany and Líšnice, and the third, south section running between Líšnice and Poněšice. The first and third sections of the fault are covered by sediments of the Pořežany Ditch (i.e. the Miocene Pořežany river), the valley of which was most likely predisposed by the fault. Several parallel lines of fault zone of the Líšnice Fault may be observed in the outcrops along the second section where the fault zone runs outside the Pořežany Ditch. The outcrops in the valley of the Budáček Creek reveal 2 to 3 m thick mylonite zones, however, without obvious indications of "young" movement (see DB-VI in Fig. 35 and Fig. 38).

The fault is of very old age. Similarly to the Vodňany Mylonite Zone, its most recent reactivation is probably of late Variscan dating.

One interpretation provided in [L. 129] claims that the fault played an important role in the formation of the Pořežany Ditch and contributed to the tectonic boundary of Miocene and Pliocene sediments in the ditch filling. This interpretation was displaced by the results of a detailed survey of the structure of the Pořežany Ditch, which included VES geoelectric profiling, drilling and the excavation of two trenches (Poněšice DB-V and Líšnice DB-IV in Fig. 35,) - for details see lit. [L. 187]. The survey results imply that the Tertiary filling is not limited generally to faults of the N-S direction. Likewise, the segmentation of the Pořežany Ditch into blocks as a result of post-Pliocene horizontal shifts was not confirmed. The filling of the Pořežany Ditch may be interconnected endwise along the entire section of the Miocene Pořežany or "North" river between Bečice and Munice, without any disruptions caused by vertical movements on the faults (for details see lit. [L. 168]).

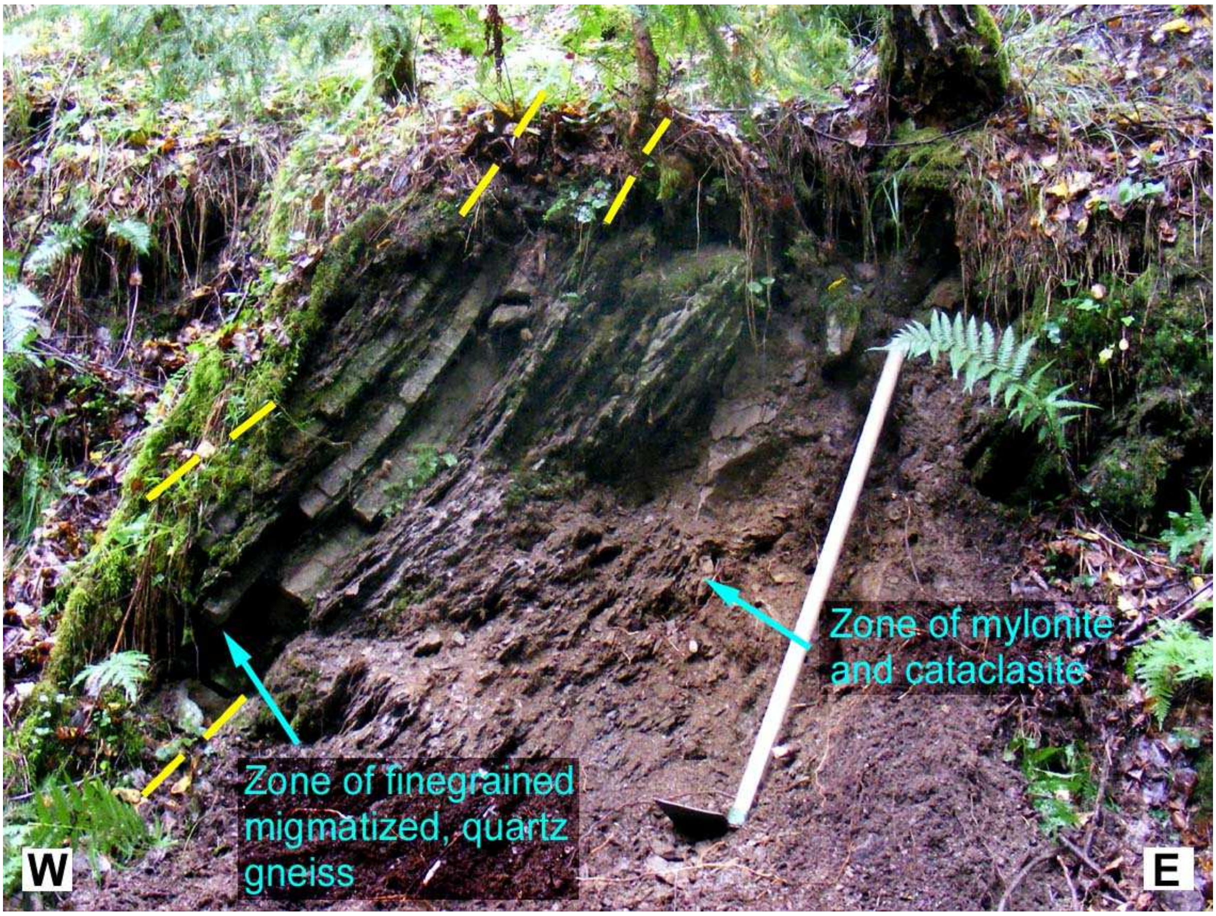

Fig. 38 View of the outcrop in the valley of the Budáček Creek where one of the lines of the Líšnice Fault may be found. In the fault zone, mylonites and cataclased biotite paragneisses alternate with more solid positions of fine-grained migmatitised gneisses. Photo by I. Prachař, 2011.

Munice Fault

Aligned in the N-S direction, the Munice Fault represents the fundamental tectonic boundary of the most deeply subsided block of sediments of the Klikov Formation in the České Budějovice Basin. No morphological manifestation of the fault is apparent in the basin flat. The course of the fault may be verified only at one location near the municipality of Munice (see lit. [L. 168]).

Geological data (namely obtained from the logging of deep drills in the České Budějovice Basin - see DB-VII in Fig. 35, see also lit [L. 168]) imply that the short section of the fault between Munice in the north and Vondrov in the south functioned as a normal synsedimentary fault in the Cretaceous. Simultaneously, the Munice Fault ends a very significant segment of the Hluboká Fault, which runs between Nemanice in the southeast and Munice in the northwest. The throw on the Munice Fault relative to the level of the basin base in the western and eastern block is about 200 m.

The Munice Fault divides the České Budějovice Basin into a deeper eastern part with a proved bounding of the basin filling by the Hluboká Fault in the north and a shallower part in the west. A subsided block of the Klikov Formation sediments, the present thickness of which reaches approximately 300 metres, may be found of the Hluboká Fault in the eastern section of the basin. The western section of the basin is shallower (around 130 m) and the sediment thickness gradually decreases to approximately 30 metres near Záblatí. West of the Munice Fault, the course of the Hluboká Fault is rather unclear and the fault does not form the tectonic margin of the basin filling.

Vlhavy Fault

In literature, the Vlhlavy Fault (N-S direction) is deemed one of the oldest dislocations in the area under review. In the 1 : 25,000 scale geological map of Netolice, it is plotted as a proved fault that runs, with minor disruptions, from Malé Chrášťany via Vlhlavy, Sedlec, Novosedly, Dubenec to Záblatí. The character of the plotted course implies that the fault bounds the Klikov Formation sediments of the České Budějovice Basin (Senonian), as well as the Mydlovary Formation sediments of the Pištín Ditch (Miocene). Thus, the fault forms the western border of the Senonian České Budějovice Basin.

Survey results [L. 167], however, have shown that the Vlhavy Fault does not follow the course plotted in geological maps (see e.g. [L. 177]). It was also ascertained that the Senonian sediments between Česká Lhota and Dubenec and the sediments of the Mydlovary Formation between Novosedly and Sedlec are not tectonically bounded with respect to the crystalline complex. The boundary between the Senonian sediments and the crystalline complex is not rectilinear; the sediments of the Klikov Formation and of the Mydlovary Formation transgress to the slightly declining slope of the crystalline complex.

New findings thus confute the interpretation accepted so far, i.e. that the Vlhavy Fault represents the tectonic boundary of the Senonian filling of the České Budějovice Basin in the west and disrupts the Tertiary sediments of the Pištín Ditch.

Until now, the existence of the Vlhavy Fault has not been disputed in literature. The interpretation of the fault was based firstly on its morphological manifestation and secondly on the coinciding direction of the fault line with the direction of other regionally significant faults. The acceptance of this interpretation was also supported by the concept of the České Budějovice Basin as a downfallen block due to faulting and by the concept of the tectonic bounding of the relics of the Mydlovary Formation sediments in and outside the basin.

With a view to the currently available data, it is not possible to completely rule out the existence of the Vlhavy Fault. The north-south direction of significant tectonic dislocations is common in the area under review. Moreover, the N-S process of the

Miocene sediments in the direction of Strachovice and Bílá Hůrka Pond could be tectonically predisposed, as well as a number of other Tertiary "ditches" (e.g. the Radomilice Ditch).

Therefore, it is not possible to rule out the presumption that the Vlhavy Fault is a part of the crystalline basement of the western closure of the České Budějovice Basin. Nevertheless, it does not form its tectonic boundary and if it runs inside the České Budějovice Basin, the potential fault throw should not be very significant (in the order of units of metres or not more than ten metres). And provided that it functioned as a synsedimentary fault, there should not be any morphological manifestation inside the basin.

N-S Blanice Valley Fault

A fault aligned in the N-S direction is plotted in geological maps[53] between the municipalities of Strpí, Čavyně and Protivín near the base of a N-S straight running slope. Further towards Tálín, the trace of the fault is plotted inside (axially) the small Miocene basin of Protivín-Žďár-Tálín. The fault is plotted as assumed and covered by younger formations. In [L. 187], this assumed fault aligned in the NNE-SSW to N-S direction that runs through the valley of the Blanice River between Vodňany and Protivín is designated as the "Blanice gap fault".

[L. 189] interprets the mentioned small Miocene basin as an erosion channel filled by fluvial-lacustrine sediments of the Mydlovary Formation in the Middle Miocene. The author presumes that the indicated fault predisposed the erosion channel.

The above specified interpretation is fairly consistent with gravity data (see lit. [L. 164]), which allow to interconnect small Miocene basins (Kloub-Chvaletice-Radčice and Protivín-Žďár-Tálín - see lit. [L. 83]) - with the coherent erosion channel filled with Miocene sediments and the Pištín Ditch in the České Budějovice Basin. This presumption was verified by a geological survey (see lit. [L. 167]). Paleoseismological investigations in the valley of the Blanice were a part of this survey and they embraced geoelectric profiling, exploratory drilling and research of the rock mass uncovered in a trench near the municipality of Číčenice (see DB-VIII in Fig. 35).

Based on the ascertained findings, it is possible to conclude that a tectonic failure zone aligned in the N-S direction undoubtedly exists and that it has predisposed the rather straight and relatively long channel of the Vodňany-Protivín-Žďár-Tálín basin. In the literature, this structure is denoted as the "Blanice gap fault". In the light of these findings, the "Blanice gap fault" seems more like a wide, tectonically weakened zone accompanied by rock fracturing than a clearly delimited fault line.

A distinct, approximately 10 m wide zone of tectonic deformation was identified by three ERT[54] profiles (Strpí 1, Strpí 2 and Číčenice). The zone generally declines very steeply to the east (see Fig. 39).

[53] 22-414 Protivín and 22-432 Vodňany.

[54] ERT (Electrical Resistivity Tomography) - A geoelectric method used within geophysical surveys, utilizing electrical resistivity tomography.

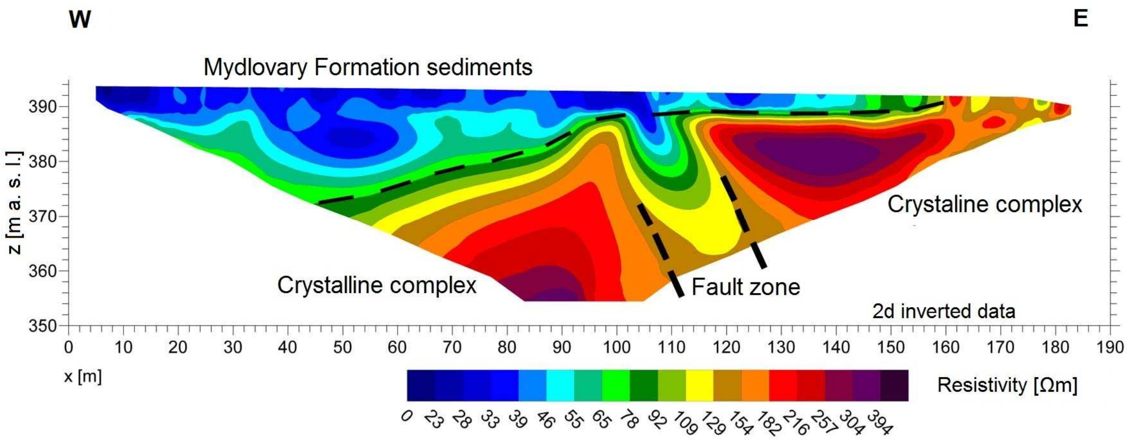

Fig. 39 Resistivity section (ERT profile) of the fault line of the Blanice valley fault near the municipality of Strpí. The section clearly shows a geophysical indication of the fault covered by an intact layer of the Mydlovary Formation sediments. Interpreted by J. Valenta. Taken from [L. 167].

The tectonic failure zone with several intrusions of younger veins of pegmatite, aplite and vein quartz was exposed by trenching in Čížčenice. Signs of older tectonic rock deformation include the presence of siliceous hornfels, pyrite, graphite along with rock tourmalinization and cataclasis.

Furthermore, the trench in Čížčenice implies that the N-S direction (the direction of the investigated "Blanice valley fault") is followed namely by joints. The course direction of tectonic failures, however, deflects in the NNE-SSW or NE-SW direction. Based on the findings uncovered in the trench, these dislocations most likely functioned as a horizontal shift.

Nevertheless, no indices were found in the trench that would point to young Pliocene or Pleistocene displacements. Likewise, the upper position of the colluvium and the geliflucted layer show no signs of failure (see lit. [L. 167]). Thus, investigations carried out in Čížčenice and Strpí did not prove any connection between the course of this tectonically weakened zone and the boundaries of the Mydlovary Formation sediments.

Hluboká Fault

The Hluboká Fault represents one of the significant fault structures in the region of Southern Bohemia. It is a system of faults aligned in the NW-SE direction along the northeastern margin of the České Budějovice Basin. According to a broader interpretation found in geological literature, the course of the fault continues southeast via the Lišov Threshold [Lišovský práh] all the way to the Třeboň Basin. To the northwest, the propagation of the fault is described as leading far beyond the western border of the České Budějovice Basin. The course of the Hluboká Fault may be unambiguously (in basic material [L. 188] designated as "Hluboká fault sensu stricto") documented in the section between Munice and Dubičné. Here, the fault constitutes the northeastern tectonic boundary of the filling of the České Budějovice Basin. The overall length of the fault defined in this manner is approximately 13 km (see lit. [L. 188]). Its hypothetical continuation outside the delimited segment to the NW and SE is substantially disrupted by fault systems aligned in the N-E or NNE-SSW direction (i.e. the Munice, Drahotěšice and Rudolfov Fault).

In the above mentioned section, the course of the fault is indicated by pronounced gravity and morphological manifestations. A marked linear maximum of the horizontal gradient of Bouguer gravity anomalies may be observed almost along the entire section of the fault between the northwestern part of Hluboká nad Vltavou and Dubičné. In [L. 188], the gradient maximum is explained as a manifestation of the steep termination of the deep part of the basin with Cretaceous sediment thickness ranging between 300 and 350 m. (SW side of the fault) relative to the rocks of the crystalline complex and the Permo-Carboniferous ones (NE side of the fault). The depth of the basin in this area is evidenced by numerous structural and hydrogeological drills, some of which also indicate a quite steep bounding of the basin.

The morphological manifestation is very pronounced particularly in the segment between Hluboká nad Vltavou and Hrdějovice and in the southern surroundings of Hosín where the slope parallel to the fault reaches the height of 65 m and the inclination of 15 to 30°, occasionally even 45°.

The survey described in [L. 188] suggests that the Hluboká Fault is not characterised by simple plane geometry with a purely linear course of the surface trace. The structure of the fault zone is complex and it consists of several segments with an azimuth ranging between 110-160°.

The Hluboká Fault system is of very old age and it is possible to presume that it has been activated multiple times throughout its geological history. The movements are envisaged as being of downthrow, as well as strike-slip character. A shift of the southwest block towards the northeast is anticipated in the above specified section of the fault.

In terms of the dating of the movements on the fault, the last significant activity is associated with the constitution of the Upper Cretaceous České Budějovice Basin. With regard to the indices obtained from the drills, e.g. the lithological development of the Klikov Formation in the proximity of the fault, we may conclude that in the Senonian, the Hluboká Fault was subjected to synsedimentary, slightly differentiated subsidence without any significant uplift of the erosive base in the largely shallow-water sedimentation area (see lit. [L. 130]).

As regards younger movements, no persuasive indices have been found to support such contemplations. The indices imply that the Hluboká Fault has experienced no significant movements since the Lower Miocene. The dating and paleoseismological evidence unambiguously confirm that no displacement has occurred on the Hluboká Fault over the period of the past 22,000 years (see DB-IX in Fig. 35 and lit. [L. 188]) - see Fig. 40. In the vicinity of the Hluboká Fault, this may be further documented by the elevation continuity of the fluvial sediments of the Vltava River from the Holocene and younger Würmian epochs in blocks separated by the fault. The configuration and elevation diversification of other terrace sediments indicates their intactness at least from the Mindelian (approx. 800,000 years).

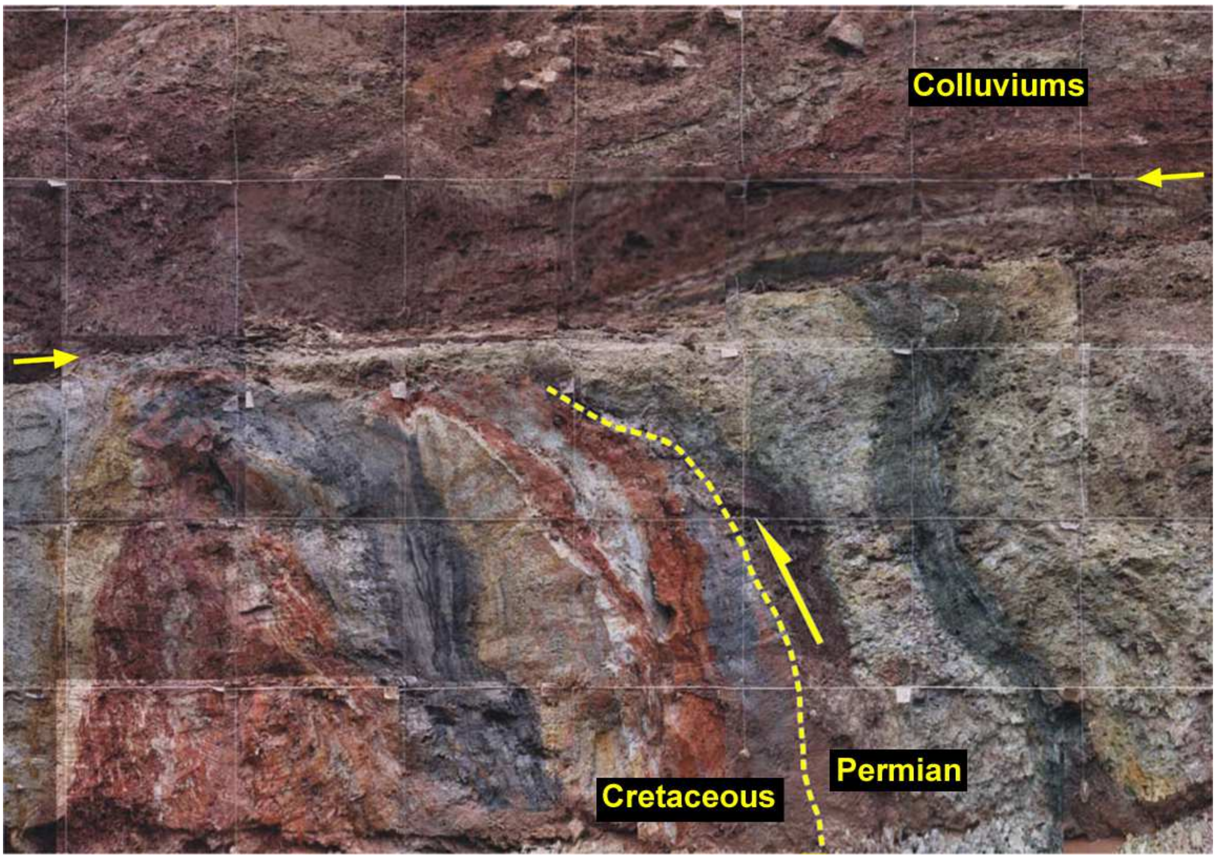

Fig. 40 A view of the wall of the R1 exploratory trench (R1 wall and a NW view) intersecting one of the branches of the Hluboká Fault in the NW-SE direction near the municipality of Úsilné. The picture clearly shows the tectonic contact of the Klikov Formation Cretaceous sediments and the Permian sediments of the České Budějovice island of the Blanice Furrow. Crushed Permian arcoses with mylonite zones (right) thrust over Cretaceous sediments in the variegated sandy-clayey facie (left). The size of one square is 50×50cm. Tectonic structures do not continue into the overlaying colluviums affected by gelifluction. The age of the intact colluvium layer was dated to approximately 22,000 years by means of the OSL[55] method. Photo by P. Špaček, 2010. Taken from [L. 188].

The continual course of Tertiary structures and sediments and the absence of tectonic movement in the post-Middle Miocene period (approx. 15 million years) may be documented in the area of Munice. The interpretation of the Munice seismic profile[56] (see DB-X in Fig. 35 and [L. 78], [L. 188]) suggests that the Hluboká Fault does not disrupt the sediments of the Mydlovary Formation that cover the fault trace between

Munice and Hluboká nad Vltavou. The profile confirms the anticipated transgressive character of the Mydlovary Formation deposition. In addition, the profile indicates that the sequence of the Miocene sediments is not disrupted by the Hluboká Fault, while the Upper Cretaceous sediments are tectonically bounded with respect to the crystalline complex. This statement may be documented by the analysis of the

55 OSL - Optically stimulated luminescence. The OSL method utilises the inner luminescence properties of minerals, i.e. the luminescence signal, which is "set to zero" when exposed to direct daylight and re-accumulated while the grain is shielded from the light. The signal is induced by natural radioactivity, e.g. produced by the ^{232}Th, ^{238}U, ^{235}U a ^{40}K radioisotopes found in many minerals or by cosmic radiation.

56 Within the AIP project (Decker, Hitersberger, Homolová (2010-2011), Austrian experts completed 3 reflexive seismic profiles in the NE part of the České Budějovice Basin intersecting the line of the Hluboká Fault (P-US profile, Úsilné, P-HO profile, Hosín and P-MU profile, Munice).

occurrence of the Zliv sandstone relics (Lower Miocene, Ottnangian), which cover the presumed trace of the fault in the surroundings of Munice (see DB-XI in Fig. 35). The trench excavated near Munice (see lit. [L. 188]) verified that the course of the Zliv layers is not disrupted by movement on the Hluboká Fault.

Investigations carried out on the Hluboká Fault as a part of the project [L. 188] did not provide evidence of any paleoseismic events in the geological record or their indications. On the other hand, numerous indices and evidence suggest that the Hluboká Fault is inactive and that its morphological manifestations are primarily the result of differential erosion in the place of contact of the slightly cemented sediments of the basin filling with the stable rocks of the crystalline complex. The intensity of erosion increased considerably in the central part of the fault due to the rapid deepening of the Vltava River.

Drahotěšice Fault

The Drahotěšice Fault is one of the major N-S faults in the near region of Temelín NPP. It belongs to the Blanice Furrow fault system and it is estimated to be of very old age. A 15-25 m thick silicified fault breccia (i.e. quartz lode) and intense rock mylonitisation may be found in the fault zone. In the course of geological history, the fault has gone through complex development characterised by multiple activations and inverse movements of the individual fault blocks. Geological literature describes its activity as an alternation of downthrow and upthrus movements accompanied by changing angles of inclination of the fault plane (for details see basic material [L. 168]).

The fault is aligned in the NNE-SSW direction. The course of the fault is best visible along the 20 km long section between Dolní Bukovsko in the north and Úsilné in the south. Nonetheless, we may presume that it continues in both the southward and the northward direction.

Between Dolní Bukovsko and Drahotěšice, the fault bounds the subsided block of sediments of the Klikov Formation with a throw ranging between 110 and 125 m as documented by drills. The fault dip is anticipated as very steep and its upturn (i.e. to the west), at least according to the logging of the ŠV-2 borehole, cannot be ruled out. In the surroundings of Drahotěšice, the course of the fault is emphasized by a quartz lode.

From Lhotice to the SSW, another parallel fault runs through the zone of the Drahotěšice Fault and forms the western boundary of the Lhotice Permo-Carboniferous basin. The Ševětín granodiorite, which forms the basement of the Permo-Carboniferous filling of the Blanice Furrow, is uplifted and forms a thin, not more than 150 m thick tectonic slice on the fault line of the Drahotěšice Fault between Chýňava and Borek.

Near the municipality of Úsilné, the course of the Drahotěšice Fault (i.e. its zone) is markedly influenced by the intersection with the Hluboká Fault aligned in the NW-SE direction. Although it is highly likely that the Drahotěšice Fault continues under the sediments of the České Budějovice Basin, a detailed specification of the fault line is complicated.

The deep ŠV-2[57] drill fundamentally contributed to learning the character of the fault zone. With respect to determining the most recent movements on the fault, the finding of one block of light-grey, slightly clayey sandstone or conglomerate surrounded by a dislocation breccia at the depth of 194.0 m and of another one at the depth of 225.6 m may be considered relevant. These rock shreds were attributed to the Klikov Formation (Upper Cretaceous) by A. Malecha who considers them evidence of inverse movements on the Drahotěšice Fault (see [L. 168]). They also bespeak of the fault activity during or after the deposition of the sediments of the Klikov Formation.

Paleoseismological investigations were carried out on an exposure of Permian sediments in a water pipeline ditch near the municipality of Úsilné (see DB-XII in Fig. 35 and [L. 167]).

Faulting of Pleistocene Deposit

The possible faulting of Pleistocene deposits, especially fluvial, was investigated at two localities. The first site is near Hluboká nad Vltavou where the river leaves the České Budějovice Basin and intersects the line of the Hluboká Fault. The second is the divide of the Blanice and Vltava rivers, i.e. the crystalline threshold between Vodňany and Zbudov.

The research of Pleistocene deposits namely involved the study of indirect indications of fault activity as direct investigations in trenches are complicated by technological difficulties, as well as a number of young geological phenomena (gelifluction, sediment re-deposition in river floodplains during flooding, etc.).

An evaluation of the first location is provided in [L. 188]. The evaluation is primarily based on a comparison of the thickness, base level and the age of the youngest (Würmian and Holocene) sediments found on the southwest (hanging-wall) block of the Hluboká Fault, in the vicinity of the weir near Hluboká, and similar sediments in the area of Hluboká-Zámostí, and the northwest (footwall) block of the fault.

The fluvial terrace sediments in the Vltava river flat that are of Würmian age according to the dating results are seated on a mildly undulate abrasion platform. As shown mainly by the results of exploratory drilling, the morphology of the platform surface is given by the lithological character of the underlying rock mass - i.e. sediments of the Klikov Formation (Senonian), the Mydlovary Formation (Miocene), and of the crystalline complex. Coarse, polymict sandy gravels with sand intercalations rest against the pre-Quaternary base. The subsequent upper layer is distinguished by a laterally and vertically variable lithological character at both investigated locations. The changes in the sediment lithology (and age) are attributable to the occurrence of Holocene flood sediments of variable thickness and grain size.

Despite the diversity of the alluvial sediments, elevation continuity at levels corresponding to the Holocene and the younger Würmian may be observed, as well as elevation continuity of the sediment basements. The dating results imply that the age of the Würmian sediments (sandy gravels, sands) is from 78 to 48 thousand years, while the age of the alluvial flood sediments is 14,700 years and less.

[57] The drill logis not available in Geofond Prague. The deep ŠV-2 borehole (depth 225.6 m) was drilled by the Central Geological Institute - ÚÚG (present-day Czech Geological Survey - ČGS) in 1986 to the northeast of Drahotěšice. The description is taken from the work of A. Malecha (1994).

A variety of investigations, dating, drilling, alluvium base modelling, e.g. utilising the results of dipole electromagnetic profiling (DEMP), were focused on the area of contact of the sediments of the Klikov Formation and of the crystalline complex. However, there are no indications of a pronounced discontinuity of the Würmian and Holocene layers that could be associated with movement on the Hluboká Fault.

Evaluations pertaining to the second investigated location (the divide of the Blanice and the Vltava in the southwestern part of the near region of Temelín NPP) were grounded on the presumption that the river systems of the paleo-Blanice and paleo-Vltava were united into a single system in the Pliocene and drained to the south. Today, on the contrary, we may see that both rivers form their own systems in the area under review, with the river divide located southeast of Číčenice, between Číčenice and Vlhavy. The causes behind the formation of the divide may have their roots in Pleistocene movements on the faults, in the updoming of the crystalline block of the current relief, or in erosion processes - in the deepening of the course of the Blanice River.

Investigations (see lit. [L. 167]) were concentrated in a part of the river divide between Vodňany and Zbudov where, due to limited catchment conditions, proluvial gravelly sands or gravels were deposited in the Pleistocene. In a geological map [L. 177], these sediments are plotted as gravelly sands and identified as $^{p}{}_{šp}Qr$ - "proluvial gravelly sands; Riss". Today, the mentioned sediments with maximum thickness of 5 m create very positive morphological elevations at levels from 395 and 398 metres above sea level.

The occurrences plotted in maps were verified and documented in the field and samples were taken for dating with the aid of the OSL method. The results of the investigation and dating of the sediments have shown that the territory between Vodňany and Zbudov functioned as a flat sedimentation area in the Rissian and Würmian periods. It was ascertained that the proluvial sandy gravels, which were found and investigated at several sites outside and along the crystalline threshold, are located at approximately the same level and are of a very similar age - i.e. around 100,000 years. In addition, moderate lowering of the base of these sediments towards the east may be observed. This is interpreted by the fact that the Blanice River had been a tributary of the Vltava River until the beginning of the Würmian. In the course of the Würmian, apparently in consequence of the deepening of the Blanice bed near Vodňany or of headward erosion by another river course, a divide was formed between the Blanice and Vltava rivers not far from Česká Lhota and Dubenec, which resulted in the separation of the two rivers.

Based on these findings, it is therefore possible to rule out the interpretation of the crystalline threshold as a tectonic horst or as a updomed part of the relief. This means that the contribution of tectonic processes to the separation of the Blanice and Vltava rivers is inconclusive, while the interpretation of the formation of the divide between the two rivers in the Würmian (100,000 to 40,000 years ago) as a result of erosion processes seems more feasible (for details see lit. [L. 167]).

Relationship of the Occurrence of Earthquakes and Known Fault Structures

An evaluation of the relationship between the occurrence of earthquakes and known fault structures was performed in an area of the "wider near region" of Temelín NPP elongated southwards to the distance of almost 70 km (see Fig. 41).

The area delimited in this manner includes two significant fault structures aligned in the N-S direction, i.e. the faults of the Blanice Furrow (3.X) and the faults of the Lhenice Ditch (4.X). The map in the figure below clearly shows that the number of foci of earthquakes and micro-earthquakes increases to the south of the near region of Temelín NPP and culminates in the area between the Danube Fault (1) and the Bavarian Lineament (2).

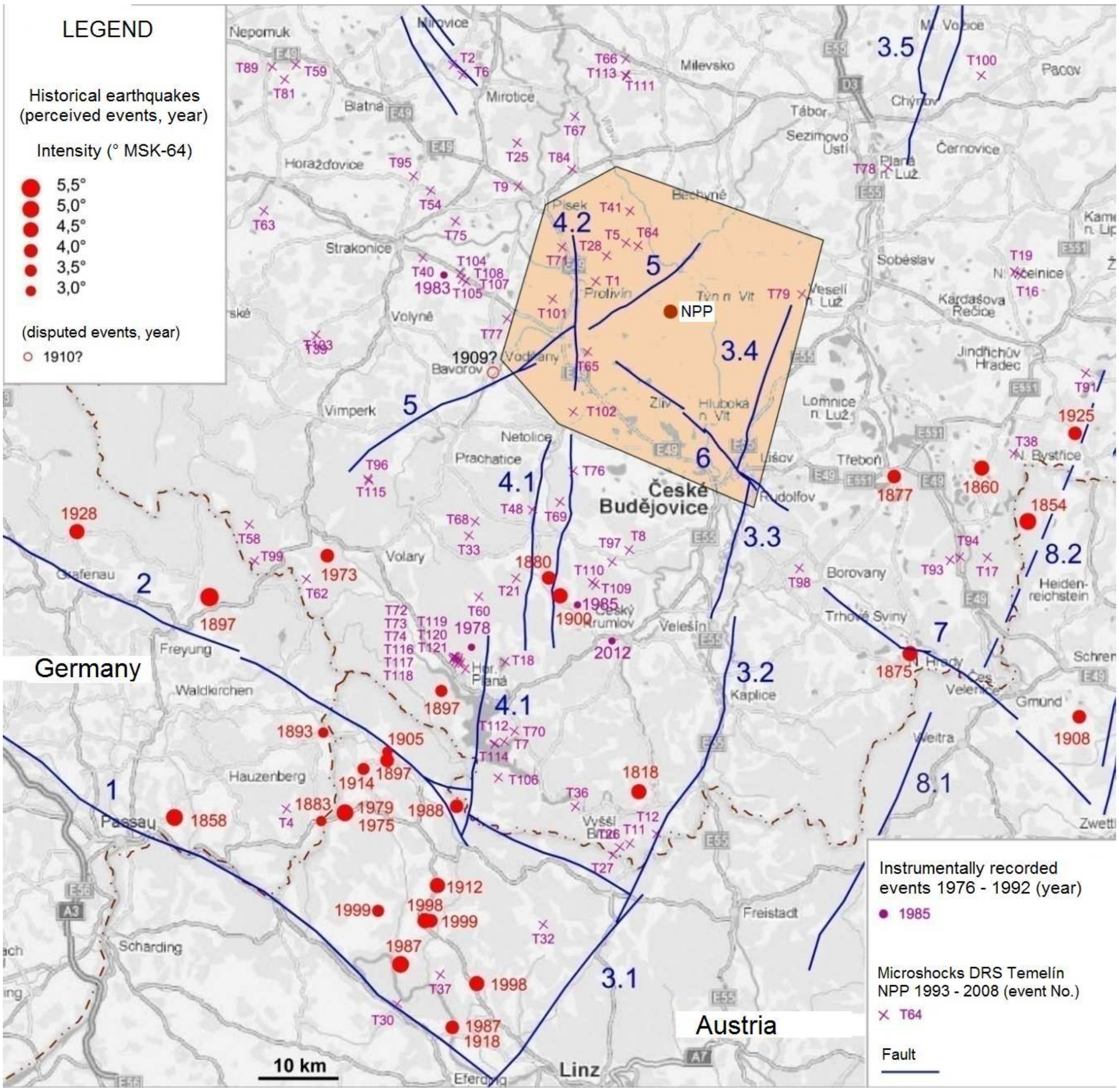

Fig. 41 The position of historical earthquakes, instrumentally recorded events and micro-earthquakes in the wider near region of Temelín NPP. The near region of Temelín NPP is delimited by an ochre background. Taken and modified from [L. 161]. Fault identification: 1 - Danube Fault, 2 - Bavarian Lineament (Pfahl), 3.1 - Rodel Fault, 3.2 - Kaplice Fault, 3.3 Rudolfov Fault, 3.4 Drahotěšice Fault, 3.5 Blanice Furrow faults, 4.1 - Lhenice Ditch faults, 4.2 – "Blanice valley fault", 5 - Vodňany Mylonite Zone, 6 - Hluboká Fault, 7 - Stropnice line, 8.1 a 8.2 - Fault of the Central massif of the Bohemian-Moravian Highlands.

The territory of the Czech Republic manifests a distinct relationship between a group of micro-earthquakes, the historical earthquake occurring on 28 May 1818 (Vyšší Brod) and the course of the Kaplice Fault in the surroundings of Vyšší Brod.

A large group of the foci of micro-earthquakes and macro-earthquakes is concentrated into a belt along the faults of the Lhenice Ditch with significant areas around Chvalšiny and Horní Planá.

A number of the foci of micro and macro-earthquakes is also linked to the fault of the central massif of the Bohemian-Moravian Highlands, which does not manifest itself on the surface of the territory of the Czech Republic.

2.6.7.2.3 Requirements According to Paragraphs 3.5 to 3.7 (NS-R-3)

The following assessment covers the territory of the site vicinity[58]. In this territory, faults plotted in official 1 : 25,000 scale geological maps were assessed, as well as geological and geomorphological indications of potential surface faulting. Fault tectonics was evaluated from the period of the Pliocene to the present.

Firstly, it should be noted that no faults are plotted in geological maps in the site vicinity, in relation to which any movement activity in the period from the Pliocene to the present is described and documented in expert literature.

Faults Plotted in Geological Maps

In the geological maps covering the site vicinity area[59], 12 verified or assumed faults were plotted in 6 localities designated by letters A to F in Fig. 42.

Locality A: Vodňany Mylonite Zone

The Vodňany Mylonite Zone is described in Section 2.6.7.2.2 to which we refer.

Locality B: Faults bounding or disrupting the Tertiary filling of the small basin of Bohunice

An approximately 700 m long fault aligned in the NWN-SES direction is plotted in the geological map (see Fig. 42) in Locality B. The fault bounds the occurrence of the sediments of the Upper unit of the Mydlovary Formation. At the same time, the fault manifests itself morphologically and an inactive fault slope is plotted in the area in the geomorphological map Fig. 43.

The fault-bounded occurrence of the Tertiary sediments of the Mydlovary Formation stems from the concept of the geological development of the region of southern Bohemia namely conceived by A. Malecha (see e.g. lit. [L. 129]). Nonetheless, a number of other geological surveys (e.g. [L. 189]) and specialised investigations and evaluations (see lit. [L. 187], [L. 188], [L. 167], [L. 168]) suggest that the Tertiary sediments are more likely remnants of the filling of the fluvial-lacustrine system rather than ditches bounded by faults. Evidence was primarily provided by the results of geological surveys and drilling carried out at several locations (near Pořežany, Líšnice, Novosedly, Česká Lhota, Čičenice - see lit. [L. 187], [L. 188], [L. 167]).

The relic of the small basin of Bohunice is interpreted in a similar manner (see [L. 168]). This interpretation is largely grounded on the evaluation of exploratory drills

[58] Paragraph 3.5 of the IAEA Safety Requirements No. NS-R-3 [L. 6] implies that the assessment concerns the area of the construction site of a nuclear installation. Paragraph 8.5 of the IAEA Specific Safety Guide No. SSG-9 [L. 14], on the other hand, defines the area as being "near the site" of the nuclear installation. In this case, the territory directly adjoining the construction site is probably considered, which is smaller than the site vicinity of the nuclear power plant.

[59] Geological maps 1 : 25,000. Pages: 22 423 Týn nad Vltavou; 22 441 Purkarec; 22 414 Protivín and Vodňany.

drilled during a deposit survey in the surroundings of Bohunice and on own geophysical survey results [L. 167]. An ERT profile was performed across the plotted fault in the direction of the slope inclination.

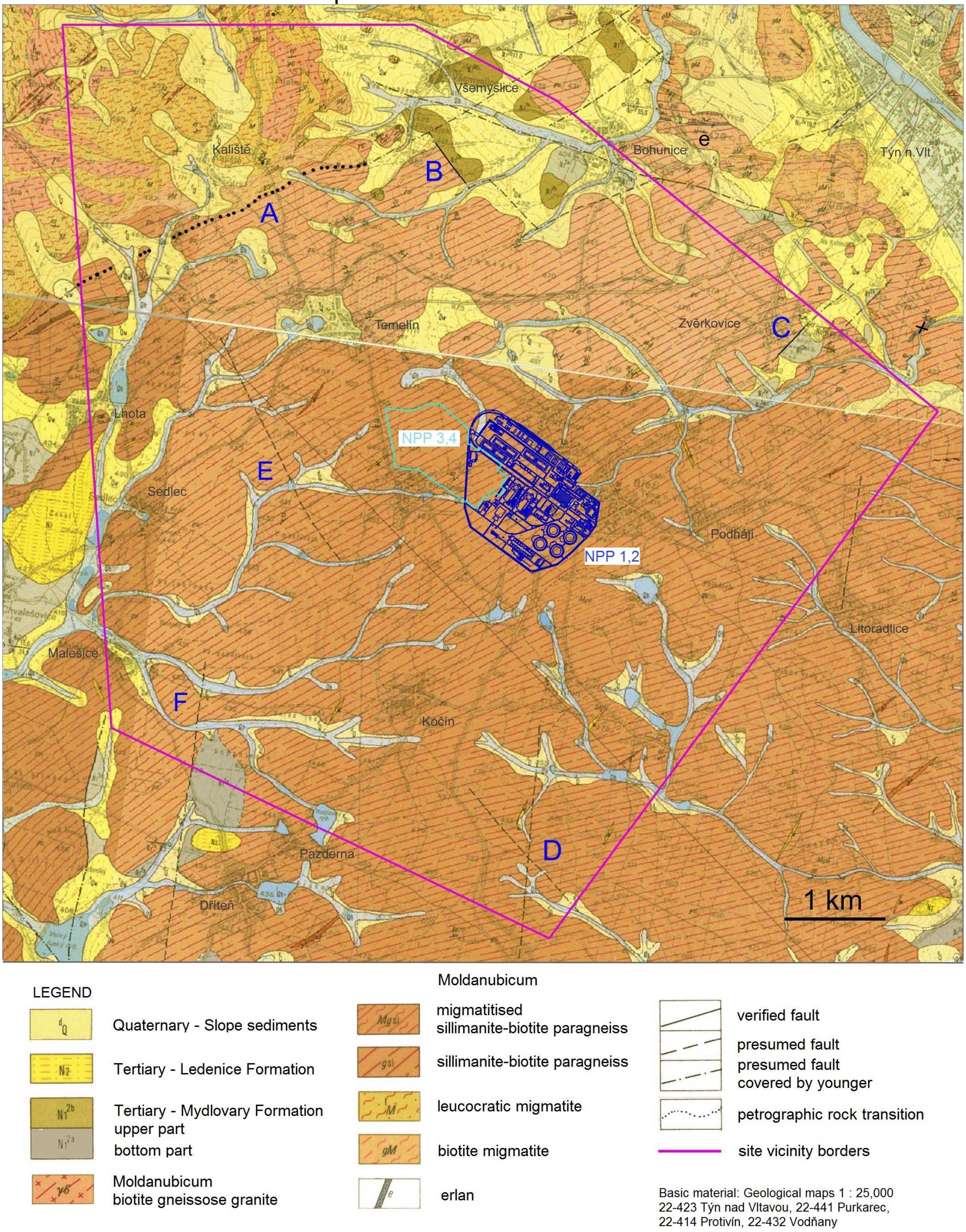

Fig. 42 Section from 1 : 25,000 scale geological maps showing the site vicinity of ETE3,4 and the localities (A to F) with plotted tectonic structures (faults). The letter "e" designates an erlan body on the Red Hill [Červený vrch] near Bohunice. The letter "x" designates the unconfirmed occurrence of erlan in the valley of the Paleček Brook [Palečkův potok] near Zvěrkovice.

The resistivity section implies that the contact of the crystalline complex with the Tertiary sediment is located 65 m further to the east and lower in the slope than plotted in the geological map. It is also apparent that the thickness increase of the small basin filling is gradual. Thus, the area of contact shows no indications of any faulting of the basin fill. The course of the slope, free of surface undulation allows to conclude that the basin sediments were not subjected to disruption caused by faults after depositing at the profile site (for details see [L. 168]).

Locality C: Faults bounding or disrupting the Tertiary filling near Zvěrkovice

According to the plot in the geological map, a part of the Tertiary sediment relic is bounded by a fault in Locality C. The presumed continuation of the fault to the northeast is covered with slope sediments and its presumed continuation to the southwest disrupts the crystalline rock mass.

The interpretation of this fault plot is consistent with the previous interpretation (see Locality B). The fault does not manifest itself morphologically in the terrain. The relief morphology was remodelled at this locality by Quaternary fluvial erosion of the Paleček Creek [Palečkův potok].

Post-Miocene tectonic activity is not apparent along the presumed southeastern continuation of the fault either. The plotted fault intersects a morphological elevation associated with the presence of a smaller body of coarse-grained pegmatite. The elevation is not disrupted by tectonics accompanied by morphological manifestations,

i.e. either by downthrow or horizontal shift (for details see [L. 168]).

Localities D, E, F: Assumed faults in the crystalline complex

Assumed faults mostly aligned in the NNE-SSW direction are plotted in Localities D, E and F in the geological map. Analogically to the faults of the Blanice Furrow, the tectonic activity of these NNE-SSW faults is late Variscan as indicated in [L. 196]. The plotted faults are based on geophysical indications - as lines ofmagnetic discontinuity (see lit. [L. 134]). Nevertheless, other evidence of their existence is not supported. In [L. 195], the author admits that a complex fold structure may give a similar geophysical picture as ruptual tectonics.

The faults have no morphological manifestations that would indicate young tectonic activity. Their course is not emphasised by preferential directions of Quaternary fluvial erosion. There are also no apparent indications suggesting that the faults are associated with a more significant lithological boundary (for details see [L. 168]).

These fault indications, if they are indications of ruptual tectonics, may be interpreted as old tectonic discontinuities of Variscan or pre-Variscan age.

Geomorphological Features of Relief

The geomorphological features of relief that could support the suspected existence of a capable fault in the period from the Pliocene to the present are shown on the geomorphological map of the site vicinity area Fig. 43.

Number "1" designates the plot of "a straight running slope of potential tectonic origin with an angle of inclination ranging from 5 to 15°" [L. 52]. The specified basic material interprets the slope as "old, inactive". Along with other slopes between the former village of Temelínec and Dříteň - see "2" - (also inclined to the west), the slope forms the western border of a peneplain with the highest elevation (the so-called Temelín Uplands [Temelínská pahorkatina]), where the Temelín NPP is situated. Along the transition to lower peneplain - Neogene lacustrine basin (the so-called Chvalešovice Uplands [Chvalešovická pahorkatina]) - pediments are deposited at the foot of the slopes in a typically concave-bent profile. The angle of the slopes gradually decrements from approximately 5° to 2° as they transform into the flat surface of the Neogene basin. These peneplains were probably diversified in the course of the Neogene phase in connection with the formation of the fluvial-lacustrine system of the region of southern Bohemia, whereas the lower peneplain of the Chvalešovice Uplands [Chvalešovická pahorkatina] was subsequently covered by Miocene and possibly also by Pliocene sediments (see rare relics). The removal of the sedimentary cover in the period from the Pliocene to the Holocene caused the exhumation of the margin of the Neogene sedimentation area and highlights the straight running slopes that, from the present-day view, appear as very young. Only relics of Miocene sediments were preserved in fillings of the deepened erosion channels of Tertiary rivers [L. 187]. With a view to the fact that the examined slopes are not associated with the course of any lithological boundary and are located at or near the presumed margin of the Miocene sedimentation area or along the lines of assumed fluvial axes, their origin is interpreted as erosion-based (for details see [L. 168]). Therefore, the slopes do not represent active tectonic relief elements.

In 1994, within specialised research aimed at evaluating the geological and tectonic structure in the area of the near region of Temelín NPP [L. 187], the area, including the close vicinity of the site area of Temelín NPP, were subjected to a geomorphological analysis. Concurrently, [L. 187] provided a comprehensive evaluation of the available geological publications, archive geological information on the locality and of 1 : 25,000 scale geological maps developed during the geological mapping of Czechoslovakia in the 1950s and 1960s.

Occurrence of Distinct Lithological Boundaries

According to geological maps as well as the results of own mapping, no distinct lithological boundaries (tectonically downthrowed blocks of mainly young Pliocene and Quaternary sedimentary rock) with or without morphological manifestations have been identified in the area of the above mentioned slopes or at other locations falling within the site vicinity (for details see [L. 168]).

Number "3" was assigned to an examined slope, which the authors of [L. 52] rated as "a fault slope markedly transformed by exogenous process with an angle of inclination from 15 to 30°", but also as "old, inactive". The inclusion among "fault slopes" was derived from the connection with the margin of the relic of Neogene sediments that were bounded by faults in geological maps. Nonetheless, according to the interpretation provided in [L. 187] and supported by investigations in the Pořežany Ditch, these are fillings of deepened erosion channels of Tertiary rivers without any bounding by faults. See also comments to Locality B above.

Occurrence of Rocks - Products of Mechanical Rock Deformation along Tectonic Lines

The relevant types of rocks and minerals found in the joints and fillings of tectonic failures were investigated namely on the construction site of Temelín NPP. Their description is provided in Section 2.6.7.2.4, as a part of the evaluation according to the criterion pursuant to Section 4(f) of Decree No. 215/1997 Coll. [L. 1].

Occurence of the Foci of Earthquakes and Micro-Earthquakes

Based on the results of seismic micro-zoning (see basic material [L. 48]), no occurrence of micro-earthquakes was detected in the site vicinity (see also map in Fig. 29).

Assessment of the Fault Reactivation Potential

The fault reactivation model allows to determine the reactivation tendency of variously oriented faults when exposed to the concurrent effect of crustal stress, i.e. through an increase in stress level, as well as a decrease in rock mass resistance (e.g. due to fluids).

The calculation of the relative reactivation potential was applied to the main faults in the wider area (i.e. the Blanice Furrow, the Lhenice Ditch, the eastern marginal fault of the Třeboň Basin and the Hluboká Fault). The selection of the appropriate faults was conformed to the knowledge of tectonic parameters (direction and inclination of the fault plane). The results may be applied to other faults of similar directions. The assessment results are presented in the form of diagrams in Fig. 44.

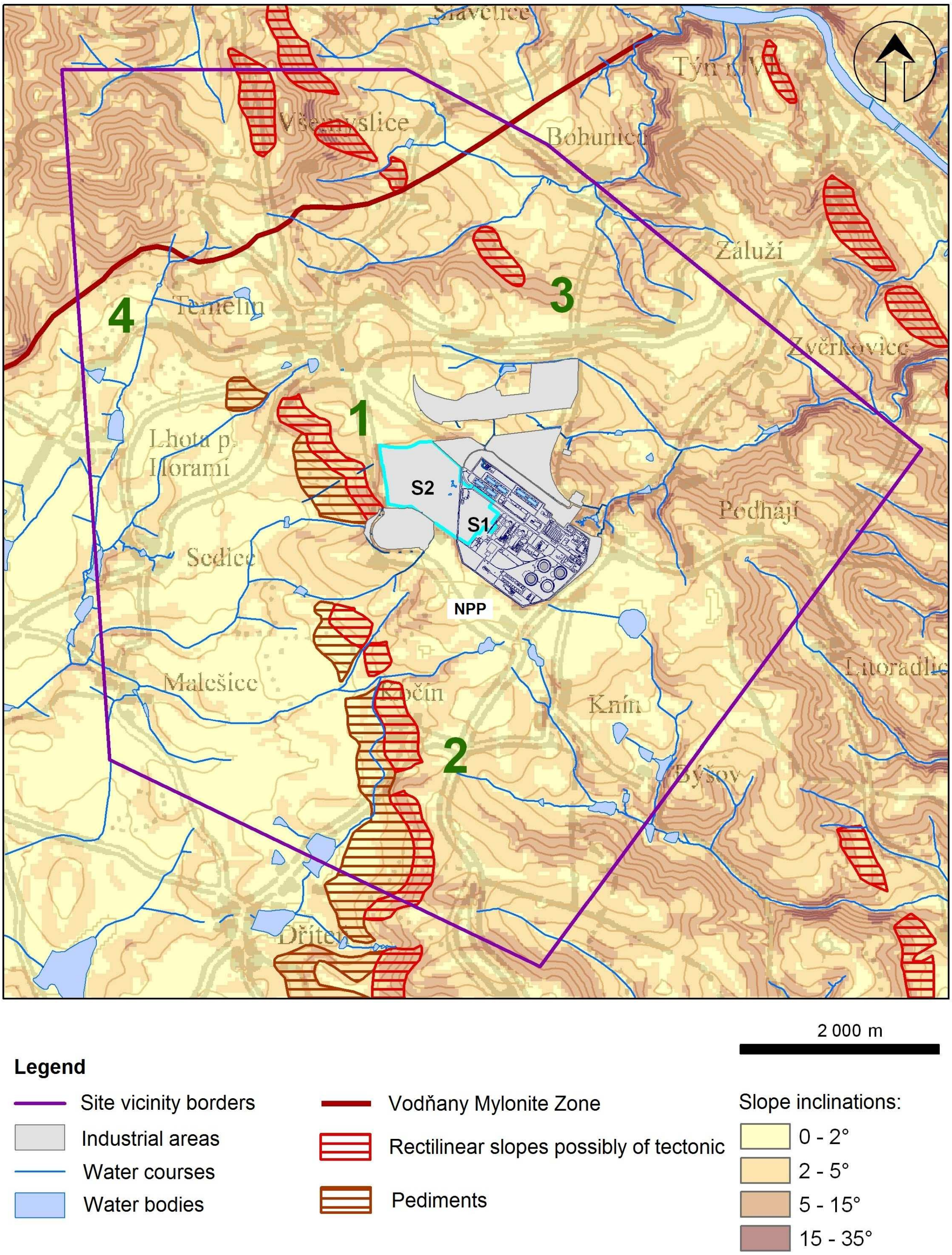

Fig. 43 Interpretation of 1 : 25,000 scale geomorphological maps (see basic material [{L. 187) showing the construction site of ETE3,4 (areas S1 + S2), the borders of the site vicinity within the radius of 3 km from the border of the land presumed for the siting and selected geomorphological and geological features of relief (numbered 1 to 4).

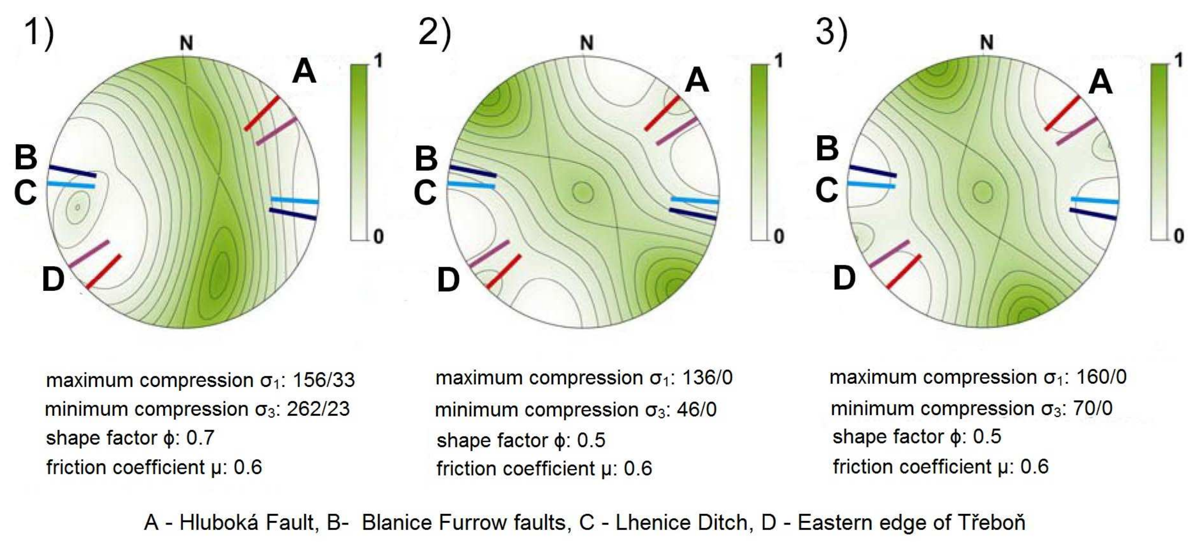

Fig. 44 Contour diagrams of the relative reactivation potential Δr for three reduced stress tensors corresponding to the stress fields determined by: 1) focal mechanisms in the epicentral area of Nový Kostel in western Bohemia; 2) overcoring stress analysis in Příbram; 3) hydrofracturing and breakout stress analyses at the KTB borehole on the southwestern edge of the Bohemian Massif. The relative reactivation potential values are plotted relative to the poles of the planes for which they were computed. The green background designates the direction with the lowest reactivation potential. Different colours were used to plot the potential poles of the fault planes directionally corresponding to the Hluboká Fault (A - red), the faults of the Blanice Furrow (B - navy blue), the Lhenice Ditch (C - light blue), and the eastern marginal fault of the Třeboň Basin (D - purple). Taken from [L. 188].

Despite the fact that the model results are influenced by the absence of findings on the parameters of the recent stress field in the near surroundings of the area under review, it still provides a sufficiently conservative estimate. The diagram clearly shows that the prevailing directions of the main fault systems in the wider near region of Temelín NPP (NNE – SSW and NW – SE) belong to directions that appear to have a higher tendency to be activated in the recent stress field. With respect to the fault of the Lhenice Ditch, the model results are fairly consistent with the findings of earthquake monitoring in the area of Chvalšiny and Horní Planá (cf. lit. [L. 92]).

2.6.7.2.4 Criterion According to Section 4(f) (Decree No. 215/1997 Coll.)

The evaluation pursuant to the criterion defined in Section 4(f) of Decree No. 215/1997 Coll. [L. 1] is primarily grounded on the results of surveys and investigations carried out on the construction site between 1979 and 1989 and between 2006 and 2010. Detailed surveys performed between the years 1979 and 1989 focused on assessing the construction site of Temelín NPP for the siting of 4 VVER 1000 units (see basic materials [L. 144] to [L. 148]). Field investigations embraced geophysical measurements, drilling, and the excavation of test pits and the R1 trench for the execution of field tests. All field geological work carried out on the construction site was documented in detail, the obtained information and data were resorted [L. 159] and they are currently available for evaluation purposes.

In 2006 and 2008, two engineering geological surveys were performed at the site of the planned construction of the spent nuclear fuel repository (see EGP repository basic materials [L. 155] and [L. 156]). Another survey was effected in two phases in areas S1 and S2 on the ETE3,4 construction site between years 2008 and 2010 (see basic material [L. 159] and [L. 165]). The survey carried out in 2010 [L. 165]) focused, among other things, on verifying the character of weakened zones (i.e. zones with a higher degree of rock weathering and jointing), which were selected based on research papers summarised in [L. 159].

Since, prior to assessing the ETE3,4 construction site, rough ground shaping had been performed and NPP1,2 had been constructed on the Temelín NPP construction site, the actual assessment was mainly based on examining archive basic materials, i.e. the documentation pertaining to drill cores and foundation pits.

Records of the potential presence of mylonite and other indications of "young" fault displacement were investigated (e.g. fault gouge) in archive basic materials. A set of descriptions of 80 core drills drilled on the ETE3,4 construction site during an engineering geological survey of the construction site of Temelín NPP was evaluated (see report [L. 144]). Various types of rock disruptions were found by geologists on the drill core and they were typically described as "tectonic failure". Nevertheless, an in-depth analysis of the descriptions has shown that many cases involved intense jointing, often spatially interconnected with quartz, vein granite or pegmatite injections. Foliation bending was also detected in the injected parts of the rock mass. The observed tectonic slickensides were usually found with limonite coating in close joints. Biotite chloritization in the joints was also frequently reported.

In the area of the construction of the reactor buildings on the Temelín NPP construction site (see basic material [L. 145]), vertical, uneven and close discontinuities of more or less N-S direction prevail. They are bound to rigid rock bodies (pegmatite, vein quartz, aplite), often mutually incoherent and occasionally almost thinned out. The same basic material suggests that no continuous or thicker "young hydrothermal filling, fault-gouge, etc." were identified in these discontinuity planes. [L. 119] describes 6 documentation points (T-9 to T-14) in relation to the NPP1,2 construction site, respectively in relation to the foundation pits of several structures. The document contains records of a number of findings of tectonic dislocations with striation, slickensiding and "tectonic clay" filling. In terms of their direction, the dislocations with fillings were oriented to the NNW-SSE or NNE-SSW with inclinations varying from vertical to an angle of approximately 65°.

Fig. 45 View of an approx. 5 cm thick filling of a tectonic dislocation in the spent fuel repository (SFR) foundation base in Temelín NPP. The filling consisted of green crumbly clay with shreds of surrounding rocks. The disturbance with the filling was approximately 10 m long and onwards thinned out. Photo by I. Prachař, 2009.

According to the expert opinion cited in [L. 168], the presence of sillimanite, authigene muscovite (sericite) and "foliation fish" in joints and on foliation planes (see lit. [L. 119]) it is possible to conclude that the observed displacements are of old age, most likely Variscan.

Similar types of rock disruptions, including slickensiding, striation and rock cataclasis, were found in the cores of newly drilled holes [L. 165] on the ETE3,4 construction site.

The evaluation of the character of the filling of tectonic dislocations (pursuant to item 3 of the indicators of "suspected" potentially capable faults - see page 375) was performed on a sample of a filling of an almost vertical dislocation of the N-S direction (see Fig. 45), found in the SFR foundation base in the site area of Temelín NPP.

An X-ray analysis of a clay fraction primarily detected (see report [L. 168]) chlorite, montmorillonite, muscovite (sericite) and secondarily quartz. The presence of chlorite in the dislocation filling indicates that it is not a product of "young" fault movements in near surface conditions.[60]

In this connection, an analysis of the data from the Temelín NPP construction site specified in [L. 119] was carried out. Unambiguous ruptures with striations were selected (see basic material [L. 168]) for the analysis (see part A). The direction of the striations on approximately 50% of the analysed planes corresponds either precisely (full arcs) or with minor deflections (dashed arcs) to reactivation by

[60] Authigene chlorite is exclusively formed under hydrothermal conditions. Formation temperatures range from approx. 150 to 300°C. Their occurrence in dislocation clays always points to older movements accompanied by increased temperatures and pressures. Thus, they are not formed in the Quaternary in low temperature and pressure conditions.

compression-based stress in the ENE-WSW direction in a normal to strike-slip regime (see part B). The results of the P and T axis[61] analysis were very similar (see part C).

The analysis implies that the determined stress is probably the record of an older tectonic phase, which manifested itself most notably through brittle failure activation at the majority of the studied localities in the surroundings of Temelín NPP. The basic stress parameters, however, do not correspond to the current crustal stress. The fact that it is an older phase is also evidenced by authigene sericite mentioned in [L. 119].

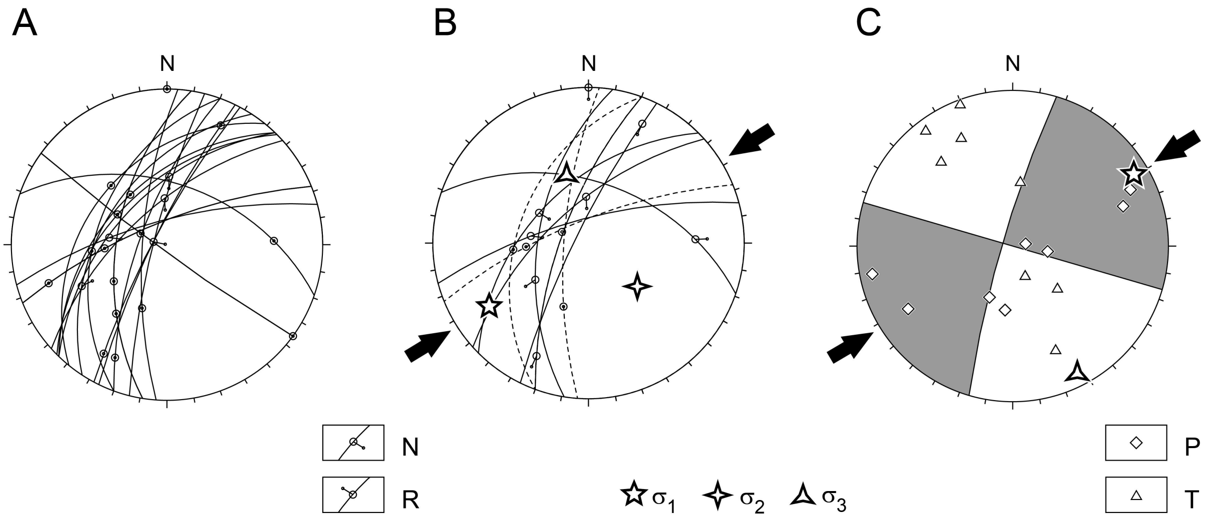

Fig. 46 Evaluation of ruptures with striations. Taken from basic material cited in [L. 168].

2.6.7.2.5 Summarized Evaluation

With a view to the above specified findings, it is possible to state that, on the ETE3,4 construction site or in its vicinity (see Paragraph 8.5 of the IAEA Specific Safety Guide No. SSG-9 [L. 14]), neither faults have been identified that would fulfil the conditions set out in Paragraph 3.6 of the IAEA Safety Requirements No. NS-R-3 [L. 6] nor have events been observed that would imply any suspected occurrence of such phenomena (see "suspicion" indications on page 350). The construction site is therefore not exposed to the jeopardy of fault displacement with surface manifestations. Likewise, no potentially capable fault was indicated in the close surroundings of the construction site that would imply the formation of secondary faults with surface manifestations.

Even if partial shifts with cross striation, slickensiding, mineral alteration or cataclased rock zones were observed in the gneiss massif, such phenomena cannot be associated with "young" (Upper Tertiary or recent) fault displacements. Considering the character of the disruptions and the nature of the mineral alternations in these zones (occurrence of chlorite), we may legitimately presume that the geological age of these syntectonic disruptions is Subvariscan[62] (Upper Cretaceous) or older.

[61] The directions of the presumed main stress on a shear fracture. The P axis is the pressure direction - 45°relative to the shearing plane in the direction of the striation. T axis is the tension direction, the lowest main stress positioned perpendicularly to the P axis.

[62] In some publications it is also denoted as the Subhercynian phase (Encyklopedický slovník geologických věd [Encyclopaedic Dictionary of Geological Sciences], 1983) - i.e.orogenic movements of the Alpine folding taking place in the Lower Senonian phase of the Upper Cretaceous.

The results of paleoseismological investigations carried out in the area of the Hluboká Fault, the western closure of the České Budějovice Basin, and in the valley of the Blanice River near Vodňany and Protivín clearly evidence that the fault present in these area are tectonically extinct and that no displacements have provably occurred on them over the past 100,000 years.

The seismological data collected in the region of Temelín NPP indicate that the foci of the known very strong earthquakes are too distant and their potential magnitude is too small to cause any landscape effects associated with such earthquakes on the construction site of Temelín NPP or in its close vicinity (cf. basic materials [L. 136] and [L. 178]).

It is thus possible to conclude that the ETE3,4 construction site is not in conflict with the exclusion criterion defined in Section 4(f) of Decree No. 215/1997 Coll. [L. 1] and that it is not subject to the requirement stipulated in Paragraph 3.7 of the IAEA Safety Requirements No. NS-R-3 [L. 6], laying down the obligation of considering an alternative site.

2.6.7.3 CRITERION ACCORDING TO SECTION 7.3

2.6.7.3.1 Specification of Hazards Indicated in Section 7.3

Section 7.3 in Tab. 115 contains only one criterion laid down in Section 4(i) of Decree No. 215/1997 Coll. [L. 1]. The criterion is relevant only if the nuclear installation embraces technologies requiring the preservation of a high degree of verticality or horizontality of a particular component. Examples mainly include requirements concerning the verticality of the reactor vessel axis or the horizontality of the turbogenerator stand.

The criterion specifically applies to surface deformations in the territory (surface inclination) initiated by tectonic activity in the site vicinity. Surface inclinations may be caused by brittle tectonics[63], as well as ductile tectonics[64]. The morphological manifestations of brittle tectonics are usually territorially limited to the surroundings of the fault causing surface deformations. These types of surface deformations are hazardous, especially if they occur in the territory of a nuclear installation construction site. The jeopardy is described in relation to the siting of a nuclear installation by the criterion defined in Section 4(f).

Thus, the criterion according to Section 4(i) primarily concerns the assessment of the hazard arising from ductile-type deformations of the rock massif as a consequence of recent movements of the upper layers of the Earth's crust. Slow uplifts, downthrows, alternating tilting, updoming and creeping (undulating) of the epeirogenic type [L. 152] occur. The very slow movements, often in the order of a few millimetres per year, are recorded by means of very precise geodetic measurements (precise levelling) and, at present, with the use of GPS methods.

As the mentioned movements affect extensive territorial units - morphostructural relief blocks - the delimitation of the position of the site vicinity (close surroundings of the construction site) relative to the borders of these blocks is crucial for the

[63] When exposed to load, the rock massif cracks and incoherent tectonic disturbances appear, i.e. ruptures.

[64] The rock massif is subject to plastic deformation and coherent tectonic structures are formed, i.e. flexures, folds.

assessment of the subject criterion. Along these borders, the recent movement gradient may be significant with respect to the hazard defined by the criterion. If the construction site (site vicinity) is situated on a uniform geological and morphostructural block, which is affected by the same degree and character of recent vertical movement, there is no or only a small jeopardy of potential inclination.

No requirement or recommendation presented in the IAEA Safety Guides explicitly applies to this issue. Nonetheless, an evaluation of recent vertical and horizontal movements in the region / near region of the nuclear installation is a part of the process of development of the regional seismotectonic model (see Paragraph 3.12 of the IAEA Safety Specific Guide No. SSG-9, [L. 14]). See Section 2.6.7.2.2.

2.6.7.3.2 Criterion According to Section 4(i) (Decree No. 215/1997 Coll.)

As indicated in Section 2.6.7.2, the presence of a capable or potentially capable fault was not evidenced in the construction site or in the site vicinity.

The interpretation of the tectonic development of the near region and its wider surroundings (the South Bohemian area of the Moldanubicum) implies that a more intense uplift of the surface of the territory occurred in the late Pliocene and the early Pleistocene, mostly along the southern edge of the Bohemian Massif, which is outside the near region of Temelín NPP. Manifestations of uneven updoming were also observed, even if at a smaller scale, in the younger Pleistocene and again outside the near region of Temelín NPP (cf. lit. [L. 82]).

Based on our presumptions, differentiated recent vertical movements, if any occur in the near region of Temelín NPP, would be very small and they would most likely affect those parts of the near region that are divided into multiple morphostructural relief blocks.

The map of vertical movement contours (see basic material [L. 187], part A, Section 6.2) clearly shows that the area with the highest contour gradient follows the northeastern margin of the České Budějovice Basin. The recent dynamics model (compare lit. [L. 64], too) of the near region of Temelín NPP implies that the relative subsidence in the České Budějovice Basin reaches the maximum of 0.3 mm per year, or 0.1 mm per year relative to the elevation with Temelín NPP. Subsidence tendencies are also manifested in the surroundings of Týn nad Vltavou and in the Třeboň Basin. Differentiated uplift tendencies of up to 0.2 mm per year may be traced in the area west of Vodňany, to the southwest of the longitudinal axis of the České Budějovice Basin. Pronounced uplift tendencies (of up to 0.3 mm per year relative to the basin centre) are apparent in the area of the Rudolfov horst east of České Budějovice.

The morphostructural classification of the near region of Temelín NPP (cf. basic material [L. 52]), as well as the recent dynamics model suggest that the Temelín NPP construction site is a part of a relatively large morphostructural block with a positive movement tendency. The block forms the highest level of the planed surface with no or only minor tectonic disruptions. Thus, it is possible to say that the construction site of Temelín NPP lies on a uniform and compact geological block (see lit. [L. 145]), which is subjected to the same degree and character of vertical movement.

When considering this criterion, it is also possible to make use of the experience from the operations of the NPP1 and NPP2 units where no technological difficulties have been encountered so far in relation to the handling of the fuel assemblies of the units in operation. Furthermore, geodetic measurements (precise levelling) of the vertical movements and inclination of the envelope of the NPP1 reactor building [L. 85].

Targeted measurements of the recent vertical movements of the surface of the construction site of Temelín NPP performed with the use of GPS methods are not available, although the Institute of Rock Structure and Mechanics of the Academy of Sciences of the Czech Republic built a permanent GNSS observatory (TEME) in Temelín NPP in 2006.[65]

2.6.7.3.3 Summarized Evaluation

Findings on the morphostructural classification of the near region of Temelín NPP, as well as experience from the operation of the NPP1,2 units show that the ETE3,4 construction site is not in conflict with the criterion stipulated in Section 4(i) of Decree No. 215/1997 Coll. [L. 1].

2.6.7.4 CRITERIA AND REQUIREMENTS ACCORDING TO SECTION 7.4

2.6.7.4.1 Specification of Hazards Grouped in Section 7.4

Section 7.4 in Tab. 116 groups together the hazards associated with the development of slope movements that may jeopardize the safety of a nuclear installation, in this case, that may affect the stability of the rock massif on the land selected for the siting. Slope movements or slope instability are mentioned in Paragraphs 3.33 and 3.34 of the IAEA Safety Standard No. NS-R-3 [L. 6]. The document imposes an obligation to evaluate the hazard in case potential slope instability is ascertained that could jeopardize the safety of the nuclear installation. Another recommendation provided in Paragraph 2.6 of the IAEA Safety Guide No. NS-G-3.6 [L. 13] classifies slope instability as a potentially unacceptable subsurface condition for the siting of the nuclear installation. Snow avalanches are also included among the phenomena that should be evaluated. Slope movements[66] are mentioned in the criteria defined in Section 4(i) and partially in Section 5(a) of Decree No. 215/1997 Coll. [L. 1].

2.6.7.4.2 Criterion According to Sections 4(g) and 5(a) (Decree No. 215/1997 Coll.) and Paragraphs 3.33 to 3.34 (NS-R-3)

The nature of slope movements is the downslope movement of materials induced by gravity, which means that their occurrence is bound to the existence of slopes, i.e. of sloping terrain. The classification of slope movements is based on two criteria, in particular the mechanism and the velocity of the movement. In [L. 139], 4 types of slope movements are defined: creep, slide, flow, and fall. When assessing the potential hazards, only the classification based on the speed of the process may be applied, which recognizes slow slope movements (creep), medium speed slope movements (slide) and rapid slope movements (flow and fall).

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[65] According to available information, the GNSS station is a part of the GeoNAS network (Geodynamical Network of the Academy of Sciences of the CR).

[66] Decree No. 215/1997 Coll. specifies: landslides, block slides, and plastic uplift of the subsurface. Landslides belong among medium velocity movements that mainly affect the weathering cover of the territory, the strength of which may be weakened by water saturation. Block slides belong among slow slope movements when more solid rock blocks sink into the plastic subsurface (clayey, silty) material and subsequently slide down the slope. Plastic uplifts are characterised by a prolapse of the plastic subsurface material into free space after being subjected to the load of the upper layer which also becomes disrupted.

Slow slope movements include movements caused by the outward forcing or folding of soft rocks. The block may slide downslope (block slides and cambering), soft rock mass or rocks with less bearing capacity in valley bottoms may be squeezed out (bulging) or plastic squeezing out of the subsurface may occur. Slow, creeping movements include various types of surface creep, which are quite frequent: e.g. hill-creep of debris and slope soils, downslope flexing or bending of beds and gelifluction (see lit. [L. 139]).

Landslides belong among medium velocity movements that mainly affect the weathering cover of the territory, the strength of which may be weakened by water saturation. The movement of hill mass occurs if the slope becomes unstable due to an imbalance between shear resistance *o*, which prevents movement, and shear stress *s*, which generates downslope movement. In expert literature, slides are further divided according to the nature of the slide plane into movement along the rotational, planar and composite slide plane (see lit. [L. 139]).

Rapid slope movements include rockfalls and all types of flows (e.g. dry flows, debris flows, mud flows, etc.). Flowing occurs when the weathering mantle is saturated by water (with the exception of dry flows). Falls (forming of talus slopes, flaking fragments and rolling blocks) are frequent on steep slopes and rock walls affected by erosion and jointing (see lit. [L. 139]).

The most frequent causes behind the development of slope movements include soil saturation due to precipitation, fluctuation of groundwater levels in slope sediments, freezing and melting of water in pores and joints, discharge of load at the slope base, load or vibrations in the landslide area of division, vegetation cover changes, unsuitable man-made interventions, as well as the effects of earthquakes. These factors may function either individually or in combination with other factors. Their effects are directly dependent on the sensitivity of rocks and soils and on the stability of the geological structure exposed to them.

The application of the exclusion criterion (i.e. rejection of the land for the siting of the nuclear installation) is limited to the condition that "the stability of the rock massif on the land selected for the siting" is endangered. This means that the mere presence of, for example, a slope as the basic precondition for the occurrence of slope movement on the land designated for the siting is not a sufficient reason for abandoning such land. The evaluation shall consider the level of hazard arising from the potential development of slope movements with respect to ensuring nuclear safety.

An obvious hazard is present when the structure is located in the area of division of the landslide (or of other types of slope movement), in which case the condition of the criterion is fulfilled and the structure foundations lose their stability. This means that the criterion shall be applied if the potentially unstable slope is situated directly on the land designated for the siting.

Likewise, the criterion shall be applied when the structure is located in the potential range of the area of accumulation of the landslide (or of other types of slope movement) and if there is a jeopardy of damage to the structure, technology or interference in the operability of the facility. In such cases, the territory within the range of the potential slope mass sources shall be evaluated. High and medium velocity movements are hazardous. On the contrary, slow movements (slow creep of weathered soil down the slope, gelifluction) present a low level of hazard (they do not constitute unacceptable conditions) since there is enough time to implement protective measures.

When conducting an evaluation in accordance with the above specified criteria, it is firstly necessary to determine whether a slope prone to sliding or - using the terminology in the IAEA Safety Guides - a "potentially hazardous slope" is present in the construction site or in its surroundings. The slope may be of both natural and artificial origin. If such a slope is present, a stability analysis should be carried out and the degree of slope stability (F_S) should be determined - see recommendations in Paragraphs 5.2 to 5.6 of the IAEA Safety Guide No. NS-G-3.6 [L. 13].

Based on an analysis of topographic basic materials and the results of terrain reconnaissance, it is possible to conclude that no "potentially hazardous slope" is located on the construction site. Within the preparation of the construction site of the current Temelín NPP, the lands designated for the siting of ETE3,4 (areas S1 and S2) were levelled to approximately 507 metres, 503 metres and 505 metres above sea level (see Fig. 47). These planar sections form the apex segments of the relief. Planar or moderately sloping terrain (with slope angles of up to 5°) of the former areas of the NPP1,2 construction site installations adjoins areas S1 and S2 in the north and northeast. The slopes in these areas were made artificially during the removal of the constructional plant structures. An industrial area of the NPP1,2 yard adjoins area S1 in the east and southeast. More dissected terrain may be found in the south and west. The Temelínec waste disposal facility with a number of artificial slopes is located in the south. In the west, the relief declines towards to lower elevation territory of the Chvalešovice Uplands [Chvalešovická pahorkatina]. The dividing planed surface levelled with the site area of Temelín NPP and areas S1 and S2 is separated from the lower planed surface by a medium-inclined slope with an angle of approximately 10°. Pediment may be found at the base section, which continues into the lower planed surface.

Natural slopes are covered with 2 to 3-metre thick slope sediments, mostly of sandy-silty character, which gradually change into cobble or silty-cobble gneiss eluviums. The former industrial areas (B1, B2, C) are covered by man-made ground of similar character. The angle of internal friction of these soils is medium to high and they are not sensitive to natural moisture changes. Area S1 and adjoining areas are relatively well drained. The industrial areas are drained by a partially functioning storm sewer system and namely by the backfills of various pipeline systems. The natural areas, including section E, are drained by melioration (see basic material [L. 179]). Therefore, it is possible to say that physical and mechanical properties of the soils along with the sufficient drainage of the Quaternary cover limit to a significant extent the sliding tendency of the slopes.

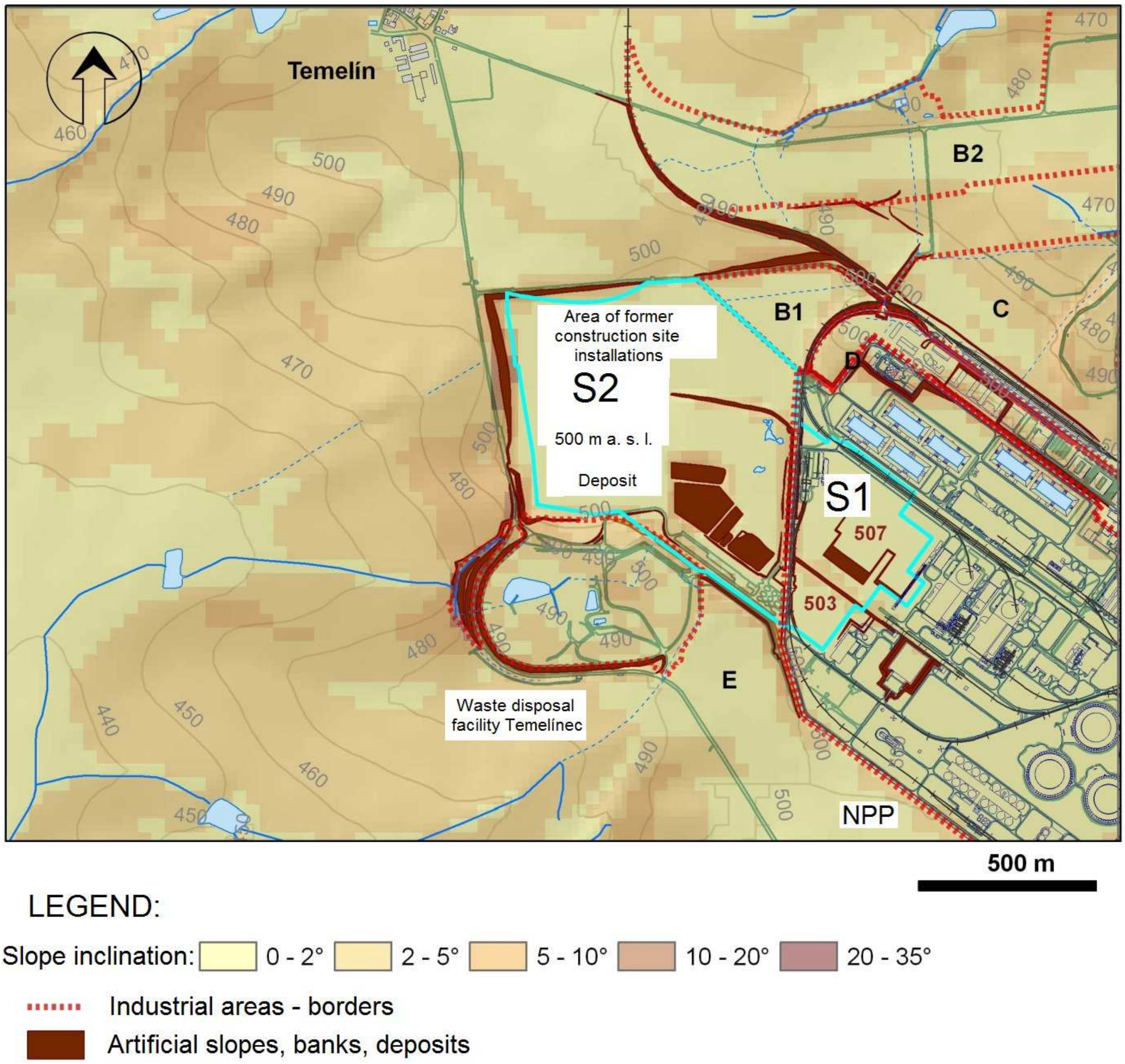

Fig. 47 Analysis of the slope angle of the relief of the land selected for the siting and its close surroundings. Elaborated with the aid of the map of Temelín NPP (ETE_34_rev002.dwg) and ZM 10 (Zabaged). Analysed with the use of the ArcGIS 10 programme.

Artificial slopes in areas S1 and S2 are not very extensive and their height is relatively low (1 to 5 m). Separate projects (including stability solution designs) were developed for surface communication bodies and siding tracks. Excavated material consisting of a mix of cohesive (silty-sandy gneiss eluviums) and non-cohesive soils (various weathered, loose gneisses) was used for their construction. The method of construction for the individual structures was pre-defined in detail.

Temporary deposits of soils and excavated material from the excavation pits of NPP1,2 structures will be removed from area S2 during the construction of ETE3,4.
The occurrence of block landslides or plastic plastic uplift of the bedrock (see wording of the criterion defined in Section (g)) is excluded in the conditions of the ETE3,4 construction site as the subsurface is not formed by fine-grained cohesive plastic soils, or by fine-grained cohesive soils that are or may become plastic, i.e. if their natural water content exceeds the plastic and/or the liquid limit.

At the same time, it is not possible to presume that areas S1 and S2 will be threatened by slow, creep slope movements after the completion of the construction due to the levelling and the draining of the area.

Considering the fact that areas S1 and S2 are located in the apex part of the relief, we may conclude that the construction site is not situated in the potential range of any mass that may be set into motion as a result of conceivable slope movements on the slopes outside the land designated for the siting. The same applies to potential avalanche hazards (see Paragraph 3.33 of the IAEA Safety Standard No. NS-R-3 [L. 6].

Based on information from the database of the Czech Geological Survey - Geofond Prague (http://www.geofond. cz/mapsphere/) - Landslides (see basic material [L. 72]), it is possible to say that no recorded "active geodynamic phenomena" mentioned in the criterion stipulated in Section 5(a) of Decree No. 215/1997 Coll. [L. 1] were identified in the site vicinity (near region) of ETE3,4.

The ETE3,4 construction site is not in conflict with the wording of the exclusion criterion laid down in Section 4(g) or the conditional criterion stipulated in Section 5(a) of Decree No. 215/1997 Coll. [L. 1]. The phenomena mentioned by the criteria do not occur on the construction site. The physical and mechanical properties of the soils, as well as other parameters of the construction site either rule out the occurrence of such phenomena (block slides, plastic uplift of the bedrock) or considerably limit the sliding tendency of the slopes. The performed relief analysis, the evaluation of the physical and mechanical properties of the soils, the analysis of the method of drainage of the sections and areas under review, as well as the terrain reconnaissance imply that no slope with the "potential for slope instability" was identified on the land designated for the siting within the meaning of Paragraphs 3.33 and 3.34 of the IAEA Safety Standard No. NS-R-3 [L. 6].

2.6.7.5 CRITERIA AND REQUIREMENTS ACCORDING TO SECTION 7.5

2.6.7.5.1 Specification of Hazards Grouped in Section 7.5

Section 7.5 in (Tab. 116) groups together the geological phenomena that present a jeopardy of collapse, subsidence or uplift of the surface in the site vicinity of the evaluated nuclear installation. Paragraphs 3.35 to 3.37 of the IAEA Safety Standard No. NS-R-3 [L. 6] impose an obligation to evaluate the site vicinity area to determine the presence of caverns, karstic formations, mines, and water or oil pumping wells. In connection with the presence of such phenomena, the potential for collapse, subsidence or uplift of the site vicinity surface shall be evaluated. Decree No. 215/1997 Coll. [L. 1] incorporates these hazards in the criteria defined by Section 4(c), Section 5(a), and Section 4(h), (n) and (o).

2.6.7.5.2 Criterion According to Section 4(c) and Section 5(a) (Decree No. 215/1997 Coll.) and Paragraphs 3.35 to 3.37 (NS-R-3)

The purpose of the criterion set out in Section 4(c) of Decree No. 215/1997 Coll. [L. 1] is to evaluate the degree of hazard arising from the occurrence of karstic phenomena on the land designated for the siting. The criterion contains the condition that the occurrence of karstic phenomena must involve the dangering of the loss of rock massif stability. This means that karsticity manifestations shall be of such level that may result in the loss of stability and thus in the decrease or loss of the overall stability of the foundation soil of the structure.

When considering this criterion, the recognition and quantification of the "extent of the karstic phenomena endangering the stability of the rock massif" are essential. Thus, attention was focused mainly on identifying the primary karstic phenomena while analysing the territory. Namely subsurface primary karstic phenomena present a jeopardy to rock massif stability, e.g. caverns, underground cavities, tunnels, karst chimneys and underground abysses, the collapse of which may result in collapse (sinking) on the surface or subsidence (formation of depressed areas). An indicator of the occurrence of such underground karstic phenomena, which may not be known or found by exploratory work, is the presence of rocks prone to karst processes (limestone, erlan, dolomite, gypsum, anhydrite) or the occurrence of above-ground primary karstic phenomena, such as karrens, sinks, chasms, collapse valleys, blind and half-blind valleys, karst pooles, swallow holes and karst springs (see also Paragraph 2.12 of the IAEA Safety Guide NS-G-3.6 [L. 13]). Another indicator is the geological and morphological position of the soluble rock stratum relative to the erosive base of the territory, and the hydrological situation in the territory (see basic material [L. 186]).

Pursuant to the criterion set out in Section 5(a), other karstic phenomena that are not specified in Section 4(c) should be evaluated in the site vicinity of the nuclear installation. The evaluation should mainly focus on verifying the presence of rock bodies prone to karsticity in the site vicinity and on documenting the manifestations of karst processes in the outcrops of these rocks, such as karrens or wider joints showing signs of karsticity.

The Monotonous Group of metamorphic rocks of the Moldanubicum, which forms the basement of the construction site of ETE3,4, rarely contains inserts of rocks capable of karsticity, e.g. crystalline limestone. If such bodies do occur, their extent is usually very small (from tens to hundreds of metres). Moreover, the inserts are enclosed in the low permeable environment of the surrounding paragneisses, which is manifested by only very limited development of karstic phenomena.

Based on the evaluation of the results of exploratory drilling on the ETE3,4 construction site (see basic materials [L. 144] to [L. 148]), it is possible to state that rocks prone to karsticity were not found in any of the boreholes. This is evidenced by the textual and graphic documentation of the cores (see basic material [L. 159]) of drills drilled up to the depth of 50 m (one drill reached the depth of 700 m below the terrain surface - see basic material L. 151]) in areas S1 and S2 at various stages of the investigations carried out on the construction site of Temelín NPP.

The occurrence of karstic phenomena in the site vicinity of ETE3,4 was evaluated by analysing 1 : 25,000 scale geological maps (see basic materials [L. 194] and [L. 135]), the "Caves and Karst Areas of the Czech Republic" map (see basic material [L. 99]) - and Fig. 48, as well as the findings obtained within terrain reconnaissance.

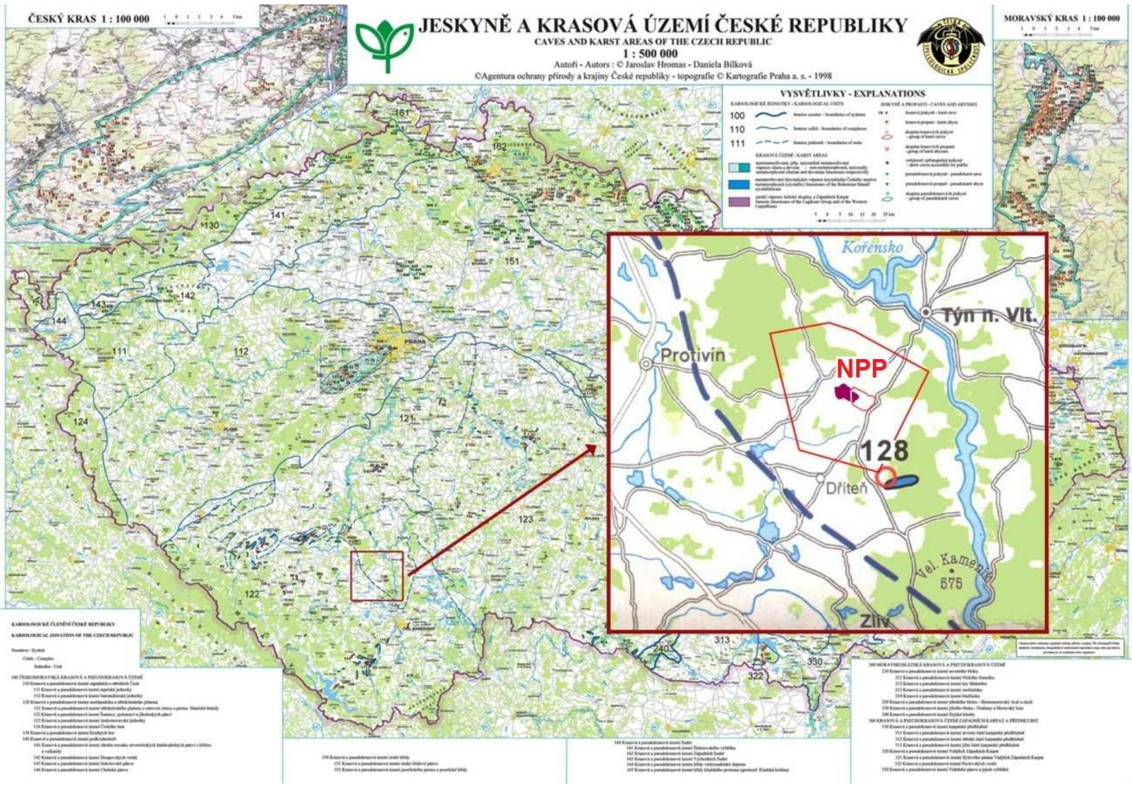

Fig. 48 Picture of the "Caves and Karst Areas of the Czech Republic" map showing the magnified section of the site vicinity of Temelín NPP (red frame), the border of the guarded area of Temelín NPP (red line) and areas S1 and S2 (dark red background). No 128 designates the position of the cave in the "Rachačka" valley. The map was taken from [L. 99].

No rock bodies prone to karsticity are plotted in the geological maps in the site vicinity zone. The closest to the border of the site vicinity zone is an erlan lens plotted east of Zvěrkovice (350 m from the border), elongated erlan lenses are plotted east of Bohunice (500 m from the border) and several lenses of erlan and crystalline limestone or erlan marble are plotted north of Chlumec (1.8 km from the border). Some calcareous rock lenses were opened by building stone quarries. A more recent deposit survey was carried out only in the locality of Chlumec (see [L. 193]).

The plotted lens near Zvěrkovice cannot be confirmed. Follow-up recognoscation did not reveal any calcareous rock outcrops or fragments in the locality.

The plot near Bohunice is an elongated erlan lens passing through the massif of the Red Hill [Červený vrch] in the E-W direction (see Fig. 42). The lens is mapped over a distance of 700 m and its thickness ranges in the first tens of metres. It was unearthed north of the top (elevation 479 m a. s. l. - Red Hill [Červený vrch]) by a shelf quarry, which largely filled with debris today. The follow-up reconnaissance of the outcrop in the western part of the quarry revealed laminated or even banded erlan-gneiss stromatite with slight signs of karsticity around joints and along foliation planes.

In terms of the extent and the intensity of the karst process, the most significant are the occurrences near Chlumec, namely the body in the Rachačka valley. The Cave Administration of the Czech Republic records three caves in this body located outside the border of the site vicinity area (see plot No. in (Fig. 48)): K1237211J00001 - Rachačka (Svatá Rozálie - St. Rosalia), K1237211J00002 - Liščí (Fox cave) and K1237211J00003 - U Výra - By Owl cave (see also basic material [L. 100]). Karstic phenomena developed into crystalline limestone rich in silicates (diopside). A detailed description of the caves unearthed in the quarry in the Rachačka valley is provided in [L. 67], [L. 105], [L. 191]. According to [L. 191], these are karstic cavities formed along almost vertical joints and largely filled with clayey soils. Secondary karstic phenomena were described in [L. 105]. Although the Rachačka cave was destroyed by quarrying between the years 1949 and 1953 (see lit. [L. 106]), relics of caves and other primary karstic phenomena are still visible in the quarries to this day (see basic material [L. 168]).

2.6.7.5.3 Criterion According to Section 4(h) (Decree No. 215/1997 Coll.) and Paragraphs 3.35 to 3.37 (NS-R-3)

With respect to this criterion, the site vicinity area is evaluated in terms of possible deformations - i.e. subsidence of the site surface in consequence of deep mining of minerals, exploitation of gas, oil and water or other substances dissolved in solutions. The criterion stipulates that the present, future or presumed deformations should be considered.

The evaluation performed in compliance with Section 4(h) of Decree No. 215/1997 Coll. [L. 1] concurrently satisfies the requirements specified in Paragraphs 3.35 to 3.37 of the IAEA Safety Standard No. IAEA NS-R-3 [L. 6], particularly in relation to Paragraph 3.35, which imposes the obligation to assess the existence of water and oil wells in the area under review.

With reference to the data obtained from the Mineral Information System (SurIS) [L. 76] and the results of the study of archive materials, it is possible to affirm that gas, oil, or water extraction or deep mining of mineral resources by their dissolution (leaching) and subsequent extraction that may endanger the stability of the rock massif in the basement of the ETE3,4 construction site have not been and are not carried out in the ETE3,4 site vicinity area. Likewise, no underground gas storages are located in the site vicinity.ETE3,4ETE3,4

2.6.7.5.4 Criterion According to Section 4(n) (Decree No. 215/1997 Coll.) and Paragraphs 3.35 to 3.37 (NS-R-3)

This criterion deals with the hazards arising from the presence of old mine workings in the site vicinity of a nuclear installation, namely in the close vicinity of the lands selected for the siting.

The proximity of old mine workings may also be a source of hazards associated with undermining (influence of underground mining), such as subsurface instability, potential collapse of the terrain (sinks), terrain deformations (formation of subsidence troughs). These hazards are assessed pursuant to the criterion defined in Section 4(h).

Old mining and extraction activities may also have other impacts, e.g. increased secondary permeability of undermined areas, formation of local depressions with draining effects, potential occurrence of intensely mineralised (aggressive) mine waters, or induced seismicity. Moreover, the proximity of old surface mining (extractive) workings or anthropogenic mining forms of the relief may put the construction site of the nuclear installation in jeopardy of development of geodynamic phenomena (slides, rock flows, mud flows from spoil banks, water breakouts due to the failure of setting pit banks, etc.).

In Paragraph 3.35 of the IAEA Safety Standard No. NS-R-3 [L. 6], the phenomena or events associated with the presence of mining activities are commented in connection with the jeopardy of potential collapse, subsidence or uplift of the surface of the site designated for the siting.

Data on old mining activities were obtained from the Map of Mining Impacts [L. 74] kept by the Czech Geological Survey – Geofond Prague.

According to the data indicated in the above mentioned map, no remnants of old mining activities may be found in the site vicinity of Temelín NPP (see map in Fig. 49).

NPP3, 4 site vicinity - Mining Impacts

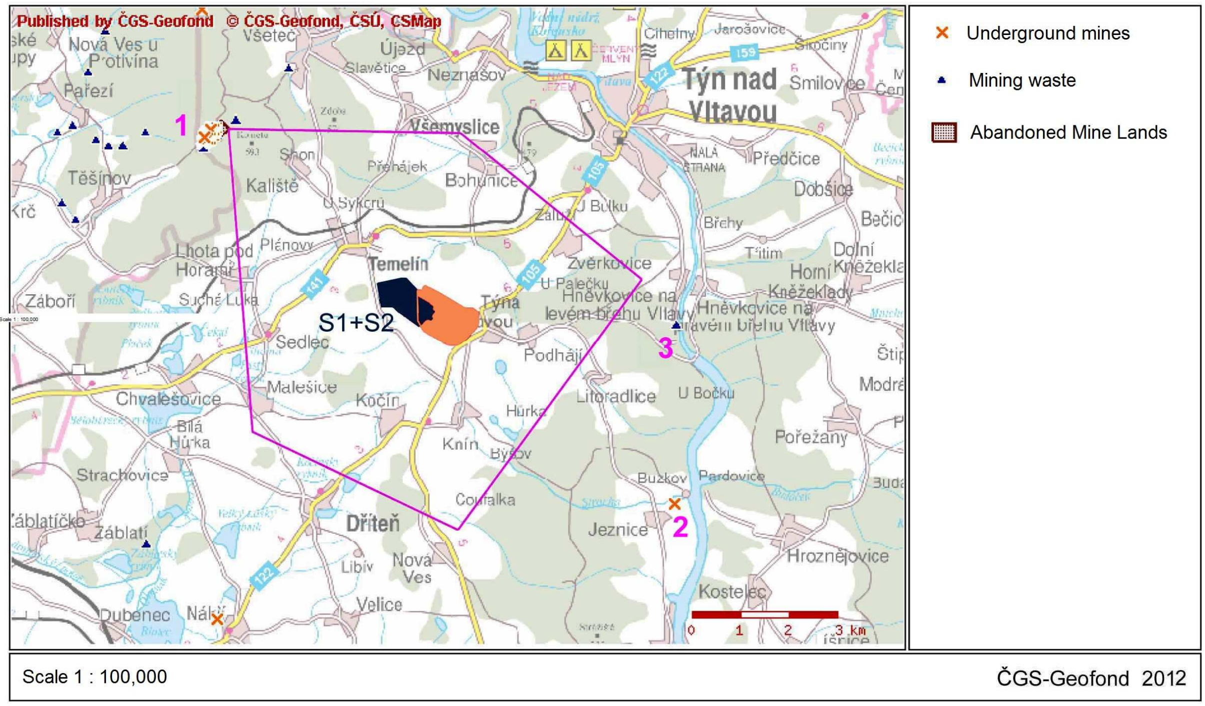

Fig. 49 Map of Mining Activity Effects

Outside the site vicinity zone (close to its borders), three notified mine workings are plotted in the map (see basic material [L. 75]). They are the remnants of the mining of gold-bearing ore at Kometa Hill near Všeteč in the period before the 16th century (see Tab. 123).

Tab. 123 Extract from the Czech Geological Survey-Geofond Prague database of "Reported Mine Workings" in the territory of the site vicinity of ETE3,4 and its close surroundings.

No.	Notification number	Year of notification	Identification	Locality	Facility type	Raw material
1	1130	2003	Všeteč Kometa - Blind shaft No. 1 (Bezedná)	Všeteč	Old mine working	Gold-bearing ore
	1543	2004	Všeteč Kometa - Blind shaft No. 2	Všeteč	Old mine working	Gold-bearing ore
	1544	2004	Všeteč Kometa - Blind shaft No. 3 (Dvojitá)	Všeteč	Old mine working	Gold-bearing ore

The remnants of gold mining at Kometa Hill are also recorded in the " Mining Impacts" database of the Czech Geological Survey - Geofond Prague (see basic materials [L. 62], [L. 71], [L. 90] and Fig. 49 - plot No. 1). Two additional sites are recorded:

Plot No. 2: Shaft for uncertain mineral in the municipality of Jeznice, dating to the 19th century (see lit. [L. 71]),

Plot No. 3: Spoil bank near Hněvkovice.

2.6.7.5.5 Criterion According to Section 4(o) (Decree No. 215/1997 Coll.)

The criterion defined in Section 4(o) imposes the obligation to assess the possible effect of ongoing raw material mining in the site vicinity on the construction of the nuclear installation and its safe operations. Similar phenomena as specified by the criterion under Section 4(n) should be evaluated. In this case, however, with respect to ongoing raw material mining.

Another purpose of this criterion, even if not explicitly formulated, is to indicate the possible conflict of interest in relation to the siting of the nuclear installation and the presence of mineral resources. It should be noted that mineral resources are subject to specific protection embedded in Act No. 44/1988 Coll., the Mining Act [L. 287]. The act lays down the principles for the protection and efficient utilization of the mineral wealth of the country. It is necessary to point out that any conflict of interest protected by the Mining Act should be resolved outside the siting process, i.e. in the spatial planning process.

The siting process considers the issue of "future" mining with respect to its potential geological effects on the nuclear installation, to which it could be exposed in the event of the commencement of mining activities in its vicinity.

Data on ongoing raw mineral mining and deposit locations (workable and out-of-balance reserves, exploration areas, etc.) were obtained from the Mineral Information System (SurIS) [L. 76] administered by the Czech Geological Survey – Geofond Prague. With respect to the exploited deposits, the following are included in the SurlS database (see Tab. 124).

Tab. 124 Overview of exploited deposits near the site vicinity of ETE3,4

No.	Deposit number	Deposit name	Mining organization	Raw material	Mining method	Opinion number
4	3139900	Bohunice nad Vltavou	Wienerberger Cihlářský průmysl, a.s., Č.Budějovice	Brick-clay raw material	Current, surface	FZ003281 - [L. 55] FZ004906 - [L. 96] FZ005864 - [L. 53] FZ006527 - [L. 140] FZ006957 - [L. 111] P069828 - [L. 137]
5	3085500	Slavětice	RENO Šumava a.s., Prachatice	Building stone	Current, surface	FZ005464 - [L. 55] FZ006893 - [L. 60] FZ006985 - [L. 56]

An overview of the mining leases and exploited deposits of non-reserved minerals is also available online in the database of the State Mining Administration at http://www.cbusbs.cz/dobyvaci-prostory.aspx or http://www.cbusbs.cz/loziska-s-tezbou.aspx.

"Bohunice I" is another exploited mining lease of brick-clay raw material plotted just outside the border of the site vicinity zone (plot No. 4 on the map in Fig. 50 and

Fišerák non-exploited mining lease of brick-clay raw material - plot No. 6). North of the border of the site vicinity lies the exploited mining lease of building stone, i.e. the Slavětice quarry (plot No. 5). Detailed information is provided in Tab. 125.

ETE3,4 site vicinity - Mining Leases

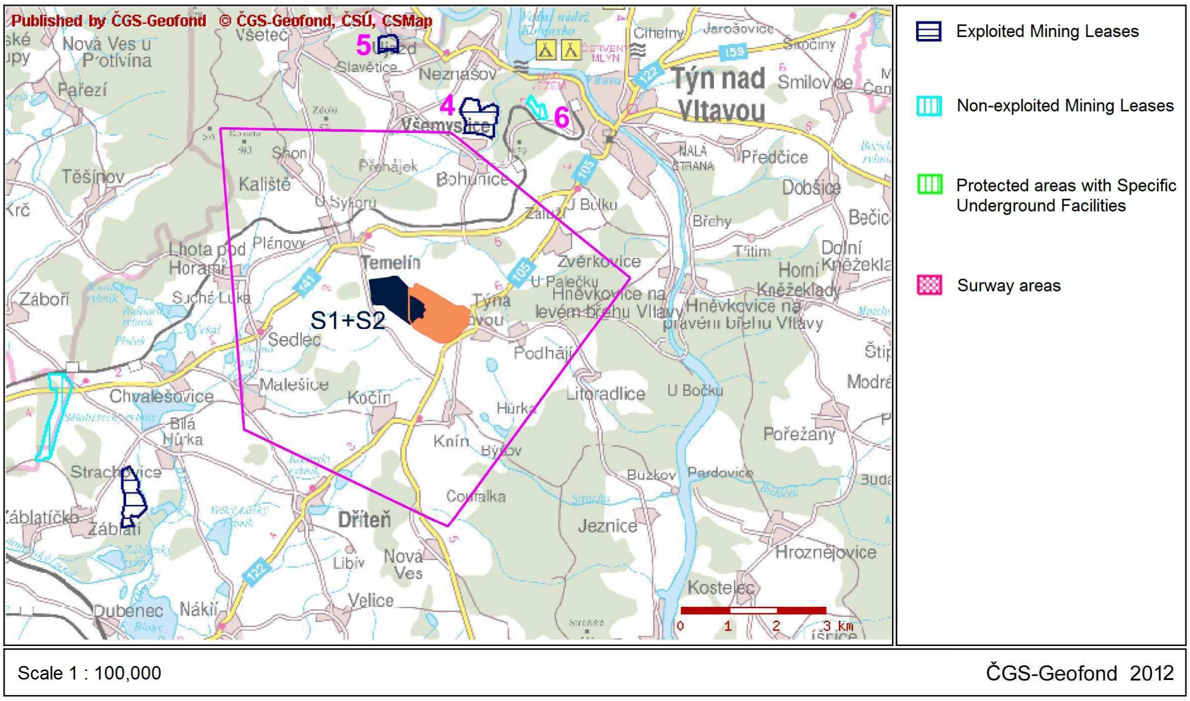

Fig. 50 Mining leases in the site vicinity of ETE3,4 and its close surroundingsETE3,4

Tab. 125 Overview of exploited allotments near the site vicinity of ETE3,4

No.	Allotment number	Name	Organization	Mineral
4	71125	Bohunice I	Wienerberger Cihlářský průmysl, a.s., ČeskéBudějovice	Brick-clay raw material
5	70716	Slavětice	RENO Šumava a.s., Prachatice	Paragneiss

Tab. 126 Overview of non-exploited allotments near the site vicinity of ETE3,4

No.	Allotment number	Name	Organization	Mineral
6	71077	Fišerák	RUMPOLD s.r.o., Praha	Brick-clay raw material

According to data obtained from the above mentioned Mineral Information System, there are no exploration areas, special protected areas, protected areas with specific underground facilities or protected deposit areas in the site vicinity of ETE3,4 (see Fig. 51). Two protected deposit areas are plotted in the proximity of the site vicinity of ETE3,4 (see Tab. 127).

Tab. 127 Overview of protected deposit areas near the site vicinity of ETE3,4

No.	Protected Deposit Area No.	Name	Company/Organization	Mineral
5	08550000	Slavětice near Všemyslice	RENO Šumava a.s., Prachatice	Building stone
7	21310 000	Týn nad Vltavou	Wienerberger Cihlářský průmysl, a.s., České Budějovice	Brick-clay raw material

ETE3,4 site vicinity - Protected deposit areas

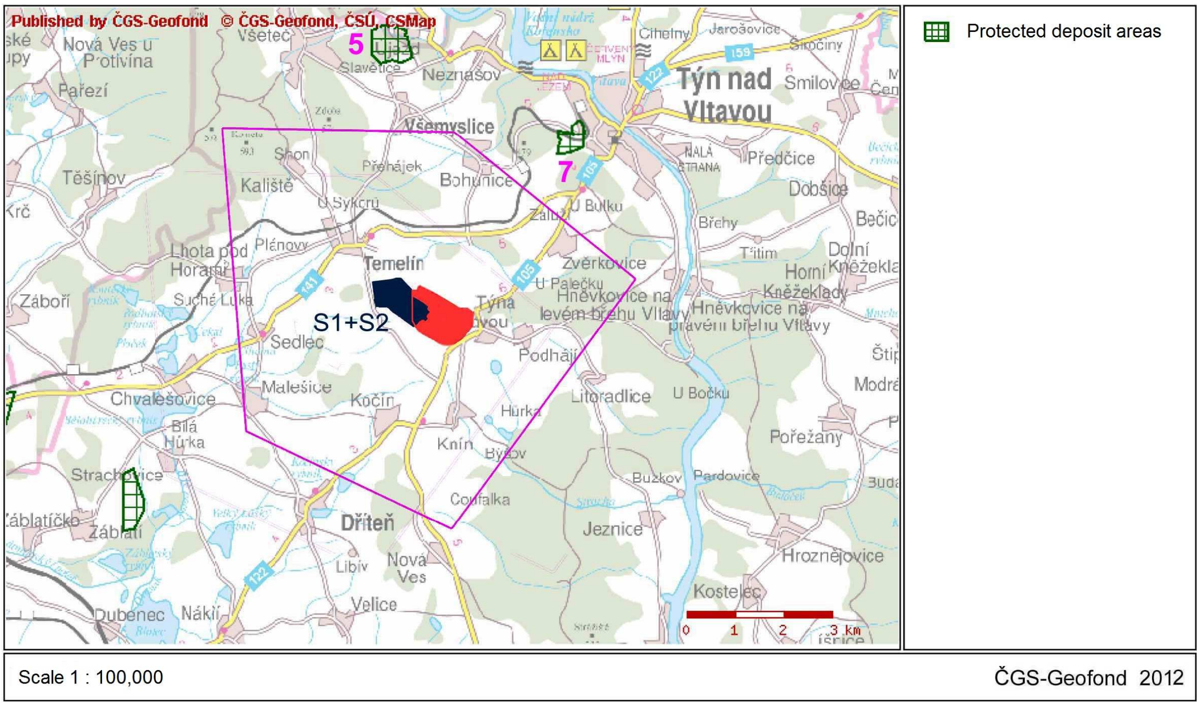

Fig. 51 Protected deposit areas in the site vicinity of Temelín NPP

In the "Mineral Deposits and Resources" subsection of the Mineral Information System (SurIS) contains the following plots (see Fig. 52). An overview of the deposits and prognostic sources is shown in Tab. 128.

Tab. 128 Overview of deposits (except for exploited) and prognostic resources of minerals in the site vicinity of ETE3,4 and its close surroundings

No.	Deposit or Resource No.	Name	Company/ Organization	Mineral	Mining Method	Geofond ID Code
6	3221500	Týn nad Vltavou-Fišerák	RUMPOLD s.r.o., Praha	Brick-clay raw material	Previous, surface	FZ006065 - [L. 58]
7	3213100	Týn nad Vltavou	Wienerberger Cihlářský průmysl, a.s., Č.Budějovice	Brick-clay raw material	Previous, surface	FZ002165 - [L. 125] FZ003203 - [L. 59] FZ005926 - [L. 97] FZ007073 - [L. 121] P021624 - [L. 77]
8	5144400	Bohunice nad Vltavou	Not indicated	Lignite	Not exploited so far	FZ003281 - [L. 55]
9	9145200	Zvěrkovice	Not indicated	Lignite	Not exploited so far	FZ003281 - [L. 55]
10	9144900	Všeteč-Karlov I.Všemyslice	Not indicated	Gravel sand	Previous, surface	P053303 - [L. 81]
11	9277400	Všemyslice	Not indicated	Building stone	Not exploited so far	P057673 - [L. 93]

No.	Deposit or Resource No.	Name	Company/ Organization	Mineral	Mining Method	Geofond ID Code
12	9120400	Všeteč	Not indicated	Gold-bearing ore	Not exploited so far	P051820 - [L. 61]
13	1012310	Chvalešovice II	Not indicated	Brick-clay raw material	Previous, surface	P055204 - [L. 185]

The map in Fig. 52 clearly shows that the siting of the nuclear installation, i.e. ETE3,4, will not lead to a conflict with the interests protected by Act No. 44/1988 Coll., the Mining Act [L. 287]. Only three plots extend to the site vicinity - namely two canceled sites of mineral deposits (plot No. 9 and 12) and one area of registered prognostic resources (plot No. 11). The potential exploitation of brick-clay raw materials near Bohunice (plot No. 8) will not have a negative impact on the construction or the operation of ETE3,4.

ETE3,4 site vicinity - Deposits and prognostic

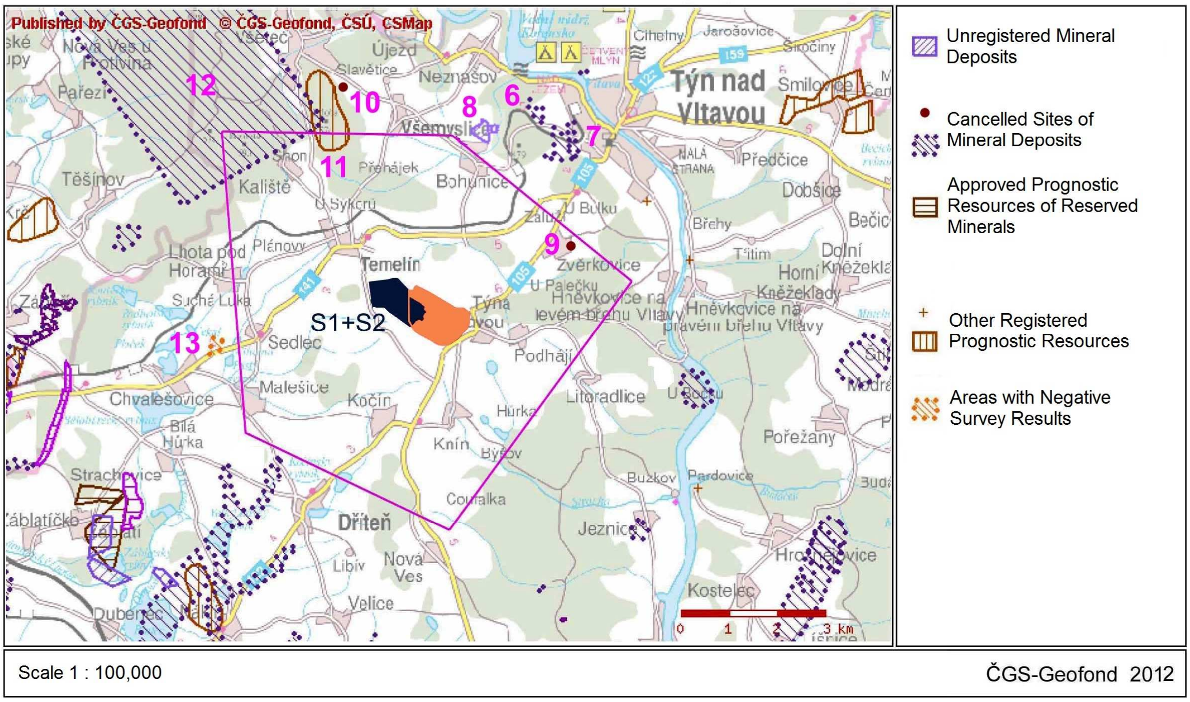

Fig. 52 Deposits (except for exploited) and prognostic sources in the site vicinity of ETE3,4 and its close surroundings

In connection with the quarrying activity at the quarry in Slavětice (plot No. 5), the possible impact of blasts (technical seismicity) on the operations of ETE3,4 was evaluated. Information and data required for this evaluation were obtained from the reports on the monitoring of micro-earthquakes in the near region of Temelín (see annual monitoring reports of the Institute of Physics of the Earth in Brno and lit. [L. 92]). These reports suggest that the "Temelín NPP Seismological Monitoring Network" registers not only natural earthquakes but also a number of artificial shocks, namely involving industrial blasting in stone quarries and planar blasting during road constructions. Approximately 20 stone quarries are located in a radius of 50 km from Temelín NPP where blasting operations of a larger scale are currently performed.

The closest is the quarry in Slavětice, located at a distance of 5.8 km. In 2011, 14 blasts were carried out at the quarry in Slavětice with maximum charge of 5 tons. The magnitude of the strongest recorded shock was ML = 1.3. According to the information provided orally by RNDr. R. Hanžlová from the Institute of Physics of the Earth in Brno, the peak ground acceleration induced by the blast on the ETE3,4 construction was estimated at 0.2 $mm.s^{-2}$. The estimation was determined by converting the value recorded by the ground motion velocity measuring device installed at the Doubravka station, which is located 5.3 km from the quarry.

2.6.7.5.6 Summarized Evaluation

Section 7.5 deals with the geological hazards involving the potential occurrence of collapse, subsidence or uplift of the surface in the site vicinity area of the assessed nuclear installation, without differentiating the nature of the initiator (natural, human activity).

Based on the performed evaluation, the following may be stated with respect to the individual relevant requirements and criteria:

[1] The ETE3,4 construction site is not in conflict with the wording of the exclusion criterion laid down in Section 4(c) or the conditional criterion stipulated in Section 5(a) of Decree No. 215/1997 Coll. [L. 1] There are no calcareous rock bodies on the construction site and their known occurrences are at locations outside the border of the site vicinity. The requirement stipulated in the applicable provision of Paragraph 3.35 of the IAEA Safety Requirements No. IAEA NS-R-3 [L. 6] has been concurrently satisfied by the conducted analyses of map data, expert literature and in-situ investigations.

[2] The ETE3,4 construction site is not in conflict with the wording of the exclusion criterion laid down in Section 4(h). The activities specified in the criterion (mining of gas, oil, water, etc.) were not and are not carried out in the site vicinity. The performed evaluation also satisfies the applicable provision of the requirement defined in Paragraph 3.35 (or 3.35 to 3.37) of the IAEA Safety Requirements No. NS-R-3 [L. 6].

[3] Old mining activities with potential consequences envisaged by the criterion laid down in Section 4(n) of Decree No. 215/1997 Coll. [L. 1] were neither ascertained in the site vicinity of ETE3,4 nor demonstrated by the study of archive materials. Likewise, the collapse of mine workings at Kometa Hill could not cause the phenomena mentioned in the above specified criterion, particularly with regard to the presumed method of exploitation of vein structures of this deposit in the surface pits and shallow jackshafts. The site vicinity therefore is not in conflict with this exclusion criterion.

[4] The ETE3,4 site vicinity is not in conflict with the exclusion criterion stipulated in Section 4(o) of Decree No. 215/1997 Coll. [L. 1] The ongoing mining of brick-clay raw materials in the deposit in Bohunice or the quarrying of building stone at the quarry in Slavětice will not affect the construction or the operations of ETE3,4.

2.6.7.6 CRITERION AND REQUIREMENT ACCORDING TO SECTION 7.6

2.6.7.6.1 Specification of Hazards Grouped in Section 7.6

Section 7.6 in Tab. 116 groups together the hazards arising from the manifestation of volcanic and post-volcanic activity in the site vicinity of Temelín NPP.

Section 4(d) of Decree No. 215/1997 Coll. [L. 1] introduces an exclusion criterion concerning the occurrence of manifestations of post-volcanic activity on the lands of the presumed siting and in the site vicinity.

Paragraph 3.52 of the IAEA Safety Requirements No. NS-R-3 [L. 6] imposes the obligation to evaluate data on phenomena that could affect the safety of the nuclear installation. Volcanism is mentioned in particular. Paragraph 2.6 of the IAEA Safety Guide No. NS-G-3.6 [L. 13] classifies volcanic activity among possible unacceptable conditions for the siting of a nuclear installation. The assessment of the hazards associated with volcanism is the subject of the newly published IAEA Specific Safety Guide No. SSG-21 [L. 110]). Nevertheless, according to the prepared guide entitled "Safety Aspects in Siting for Nuclear Installations" (DS433 - [L. 109]), this assessment should be performed in cases when the potential site of the nuclear installation is situated in volcanic regions (see Article 13, Appendix A).

2.6.7.6.2 Criterion According to Section 4(d) (Decree No. 215/1997 Coll.) and Paragraph 3.52 (NS-R-3)

An evaluation pursuant to this criterion includes dealing with the relevant aspects defined by the requirement in Paragraph 3.52 of the IAEA Safety Requirements No. NS-R-3 [L. 6].

The purpose of the criterion is to identify and assess the hazards associated with the manifestations of post-volcanic activity as a significant indicator of "recent" volcanism in the area under review. With a view to the conditions of the geological structure of the territory of the Czech Republic, volcanic processes taking place in the period from the Upper Tertiary to the older Pleistocene are considered as "recent" volcanism.

In the Czech Republic, the occurrence of apparent manifestations of recent volcanic and post-volcanic activity is limited to two areas in western and northwestern Bohemia and to the Low Jeseník Mountains [Nízký Jeseník] in Moravia. Volcanism in these regions pertains to the 4th neovolcanic phase of the Bohemian Massif based on the classification indicated in [L. 122]. Its age is estimated between 2.7 (or 3.4) - 0.86 million years (see [L. 122]). Western Bohemia is known for the occurrence of cones of slag pyroclastic rock and sodalite-melilite nephelinite, namely in the area SSW of Františkovy Lázně (Komorní hůrka) and SE of Cheb (Železná hůrka). Distinct manifestations of post-volcanic activity include CO_2 emissions at the Soos peat bog near Františkovy Lázně and thermal mineral water springs in western Bohemia. In the Low Jeseník Mountains between the towns of Leskovec nad Moravicí and Bruntál, four stratovolcanoes with lava flows (olivine basalt - nepheline basanite) and pyroclastic rock positions may be found, i.e. Small Roudný [Malý Roudný], Big Roudný [Velký Roudný], Venus Volcano [Venušina sopka] and Uhlířský Hill [Uhlířský vrch]. In addition, CO_2 emissions may be found near Bruntál, Krnov and Rýmařov.

The application of the criterion is closely bound to volcanic terrain, which may be identified by means of the ascertained volcanic shapes of the relief and by the occurrence of post-volcanic phenomena. Another sign is increased heat flow (see [L.

64] and Fig. 53). Post-volcanic phenomena embrace gas and steam emissions - mud volcanoes, mofettes[67], solfataras[68], geysers and hot springs[69]. Certain types of mineral water springs may also be associated with volcanism[70]. These are medium or highly mineralised waters (i.e. with a more than 500 mg.l^{-1} content of dissolved solid substances), carbonic and sulphurous[71]. Since mineral waters are credited for their balneological effects, this fact may serve as an additional differentiating feature.

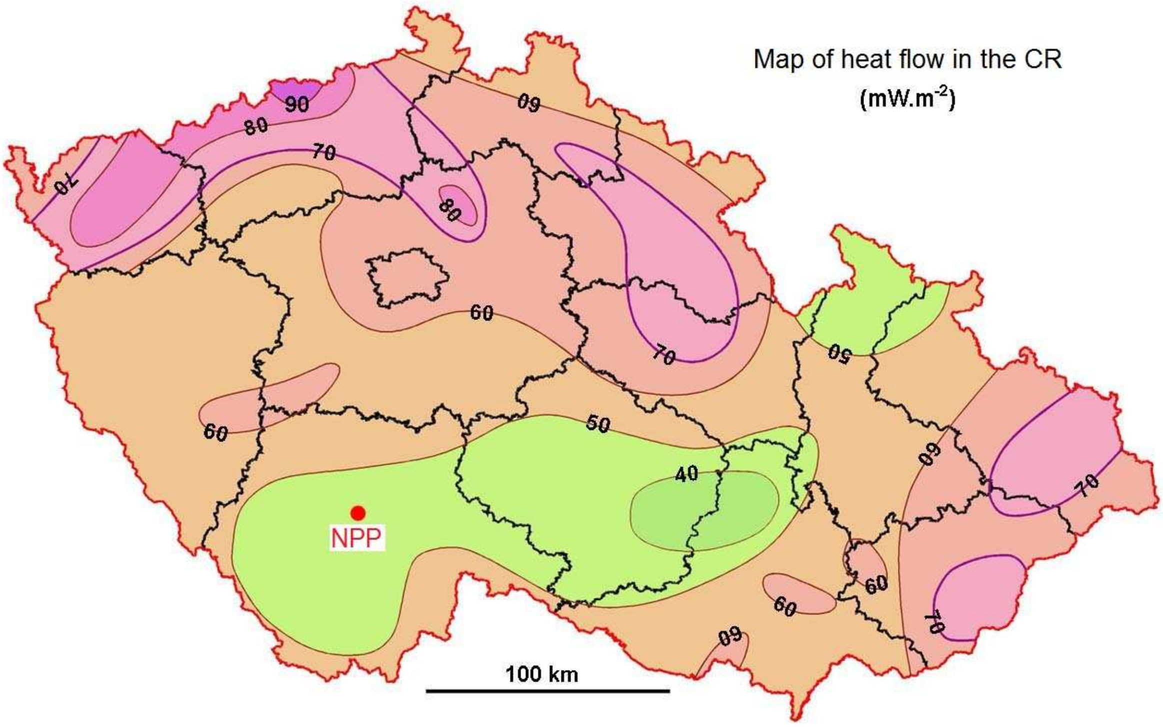

Fig. 53 Map of heat flow on the territory of the Czech Republic. Taken and modified from figure 4.1 in [L. 64].

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[67] Mud volcanoes are cones consisting of mud (intensely saturated clay). Mud volcanoes are formed by mud ejections, for example, by gases of post-volcanic character at the site of their emission (see Encyklopedický slovník geologických věd [Encyclopaedic Dictionary of Geological Sciences], 1983). Mofettes are exhalations of dry carbon dioxide at a temperature below 100°C (Encyklopedický slovník geologických věd [Encyclopaedic Dictionary of Geological Sciences], 1983)

[68] Solfataras are exhalations of vapour and gases at temperatures between 100 and 200°C, consisting of water vapour, hydrogen sulphide, sulphur dioxide, and carbon dioxide (Encyklopedický slovník geologických věd [Encyclopaedic Dictionary of Geological Sciences], 1983).

[69] Pursuant to Annex 1 of Decree No. 423/2001 Coll., "thermal waters" are natural waters the natural temperature of which at the spring site exceeds 20°C. Waters with temperatures below 20°C are designated as cold. Groundwaters are further divided according to their temperature into tepid (20 - 35°C), warm (35 - 42°C) and hot (over 42°C).

[70] From the point of view of the above specified decree (footnote 69), "mineral" water is any groundwater of original purity and properties (see also [L. 112]). "Mineralised" water mentioned by the criterion cannot be considered as an indicator of post-volcanic activity because according to the currently accepted definition it is water "containing a higher volume of dissolved substances, but it is not water of natural origin (e.g. mine water)" - (see [L. 112]).

[71] This means that the content of carbon dioxide is more than 1 g per litre of water, or the content of titratable sulphur is more than 2 mg per litre of water (Decree No. 423/2001 Coll., as amended by later regulations, Annex 1).

When performing an evaluation according to the indicated criterion, verifying the potential occurrence of post-volcanic phenomena and/or their description and quantification is essential.

The geological structure of the Šumava Moldanubicum and its geological development in the Tertiary and Quaternary (see text in introductory Section 2.6.2.2.1), completely rule out the possible occurrence of the phenomena considered by the criterion in the site vicinity of Temelín NPP or in its wider surroundings. This claim may be particularly supported by the position of Temelín NPP on the map of heat flow with values ranging from 40 to 50 $mW.m^{-2}$ (see Fig. 53).

Furthermore, it is possible to confirm that there are no known occurrences of Tertiary of Quaternary volcanic rocks (see geological map of the Czech Republic, basic material [L. 104]), or of any of the above named indicators of post-volcanic activity, including recent emissions of thermal waters, in the site vicinity zone of Temelín NPP. The map of mineral waters of the CSSR (basic material [L. 84]) indicates that there are no springs (sources) of mineral waters with parameters defined in footnote 71 in the site vicinity.

2.6.7.6.3 Summarized Evaluation

The ETE3,4 construction site and site vicinity are not in conflict with the specified exclusion criterion set out in Section 4(d) of Decree No. 215/1997 Coll. [L. 1]. The phenomena and rock types detailed in the criterion are not present in or on the evaluated territorial units.

Moreover, an assessment of the hazards arising from recent volcanic activity within the meaning of Paragraph 3.52 of the IAEA Safety Standard No. NS-R-3 [L. 6] was conducted.

2.6.7.7 CRITERIA AND REQUIREMENTS ACCORDING TO SECTION 7.7

This section (see Tab. 116) groups together the hazards arising from unfavourable geotechnical parameters of the foundation soil on the construction site of a nuclear installation. The obligation to collect geotechnical data on the foundation soil is imposed by Paragraph 3.41 of the IAEA Safety Requirements No. NS-R-3 [L. 6], and Paragraph 3.42 thereof lays down the obligation to evaluate the stability of the foundation soil under static and dynamic load. The criterion provided in Section 4 of Decree No. 215/1997 Coll. [L. 1] instructs to exclude a construction site where the bearing capacity of the foundation soil is insufficient. The conditional criterion set out in Section 5(b) of Decree No. 215/1997 Coll. [L. 1] deals with the hazards arising from the unfavourable geotechnical properties of the foundation soil.

In contrast with Decree No. 215/1997 Coll. [L. 1], the IAEA recommendations do not address these properties by any exclusion criterion. The only exceptions are soils with potential for liquefaction (for details see Section 2.6.7.8.2).

Section 2 of the IAEA Safety Guide No. NS-G-3.6 [L. 13] contains recommendations related to the analysis of the seismic response of the subsurface. The basic site categorization is determined with the aid of the velocity of shear waves in the subsurface materials. Paragraph 3.1 defines the V_S velocity limit values for site types 1 to 3. Paragraphs 3.6 to 3.14 contain detailed recommendations for identifying basic seismic parameters of the bedrock. Section 4 of the IAEA Safety Guide No. NS-G-3.6 [L. 13] presents a number of recommendations that concern the assessment of the foundation conditions, as well as the unfavourable properties of the foundation soil. The text in this section implies that the correct procedure for site evaluations is the following:

- Rocks and soils showing unfavourable foundation-related properties should be investigated in detail and monitored;
- Unfavourable properties of the foundation soil should be considered during the selection of the foundation method and construction.

2.6.7.7.1 Requirement According to Paragraph 3.41 (NS-R-3)

In addition to the obligatory determination of the geotechnical properties of the foundation soil, Paragraph 3.41 of the IAEA Safety Requirements No. NS-R-3 [L. 6] also imposes the obligation to develop a foundation soil "profile" in a form suitable for design purposes. Pursuant to Paragraphs 3.3 to 3.5 of the IAEA Safety Guide No. NS-G-3.6 [L. 13], the profile should include the following:

- Description of the rock massif, data on stratigraphy and the number of rock layers and their thickness;
- Description of the physical and chemical properties of rocks that are used for their classification;
- Determination of S and P wave velocity, a description of the deformation and strength properties of rocks, consolidation, permeability and other mechanical properties obtained by laboratory or in-situ tests.

The geotechnical data for the ETE3,4 construction site were acquired in several investigation phases described in [L. 144] to [L. 150]. The identified uncertainties were reviewed in the course of investigations by gradual densification of the network of exploratory sites and by verification surveys conducted in the Unit areas of the planned VVER 1000 units No. 3 and 4. Extensive exploratory drilling survey was carried out on the construction site along with a number of in-situ and laboratory tests of both rocks and soils. The aim of the laboratory tests was to determine the descriptive, physical and mechanical strength or deformation characteristics of the rocks and soils that were necessary for their classification.

The complementation of the information on the foundation soil in the extent required by the valid IAEA Safety Guides was executed in 2010 (see basic material [L. 165]). This material characterises the engineering-geological conditions in areas S1 and S2. In addition, an evaluation of the spread and the depth of anthropogenic layers (backfills) along with the detection of buried structures was carried out in area S1.

The foundation soil profile is detailed in Section 2.6.2.3.4 to which we refer.

The section also includes data describing the individual layers of the foundation soil profile. The indicated data were obtained through statistical evaluations of the results of laboratory rock and soil tests, in-situ tests, as well as geophysical measurements and logging.

2.6.7.7.2 Requirement According to Paragraph 3.42 (NS-R-3)

Investigations, field or laboratory tests and geophysical measurements demonstrated the stability of the foundation soil under static and dynamic load. Conditions for the execution of the foundations of structures of the 1st and 2nd seismic category on rock material exhibiting a low degree of weathering and minor signs of tectonic disruption, with a modulus of deformation of not less than 100 MPa and shear (S) wave velocity higher than 1100 $m.s^{-1}$.

The stability of the foundation soil under static load may be presented on a map of contours, showing the elevation above sea level for the achieved modulus of deformation of E_{def} = 100 MPa (Fig. 24, see also [L. 165]). The stability of the foundation soil under dynamic load is determined by a set of parameters of the subsurface rock, such as S and P waves, Young's modulus or shear modulus, as indicated in Tab. 112 in Section 2.6.2.3.7

2.6.7.7.3 Criterion According to Section 4(k) (Decree No. 215/1997 Coll.)

This criterion exclusively concerns the quality of the foundation soil, i.e. to the area of the construction site. It defines the limit values of selected engineering-geological parameters of the foundation soil, such as:

Bearing Capacity

The foundation soil bearing capacity is defined as the maximum stress under the foundation base, which does not induce plastic deformations or the formation of slide planes [L. 153]).

In general, it is possible to conclude that, based on empirical experience[72], fine-grained soils of all classes and symbols and of soft or firm consistency, certain sandy soils (particularly clayey sands or silty sands) and some gravel soils (namely clayey gravel) do not meet the requirement (0.2 MPa and less) defined by the criterion. It is possible to presume that soils with this bearing capacity have the modulus of deformation equal to E_{def} = 8 - 10 MPa or lower.

Collapsibility

Collapsibility is induced by changes in the soil structure, namely due to a collapse of the structure as a result of flowing water, vibrations, or excessive load. Collapsibility is described by the "collapsibility coefficient" I_{mp} (I_{mp} > 1 % is usual with collapsible soils). Soils with porosity $n > 40\%$ and water content $w < 12\%$ are considered collapsible.

Collapsibility is a property typically manifested by loess and loess soils. They are characterised by a grain size distribution curve that is very steep in the silt particle field and partially extends to the fine sand field. The loess macroporous structure collapses and its volume suddenly decreases due to moisture penetration. Another essential sign of proneness to collapsibility is dry bulk density lower than 1500 $kg.m^{-3}$. Collapsibility may be also observed in coarse-grained soils under dynamic load, frozen soils subjected to increasing temperatures or in clays exposed to decreasing water content.

[72] Empirical experience in the area of the foundation of structures were aptly included in the no longer valid Czech Technical Standard: CSN 73 1001 "Foundation of Structures, Foundation Soil under Shallow Foundations". Annex 6 contains "Tabular Design Values of Bearing Capacity R_{dt} of Soil for Individual Soil Classes, Depth of Foundation and Foundation Width". The standard also divides soils into classes and according to symbols. CSN 73 1001 was invalidated as of 1 April 2010 and replaced by CSN EN 1997-1 (731000) "Eurcode 7: Geotechnical Design - Part 1: General Rules".

Swelling

Swelling of soils is caused by higher shares of clay minerals, especially montmorillonite. Montmorillonite belongs among three-layer dioctaedric aluminosilicates and its important characteristic is the variable value of the *c* lattice constant, or the ability to change the intralayer distance depending on the moisture volume or the size of the cation or complex bound in the intralayer space. In practice, this property manifests itself in the ability to expand, i.e. to grow in volume as a result of moisturization.

The swelling of soils is determined by oedometer tests.

Organic Content

The criterion limits the content of organic matter in the foundation soil to 3% (of dry soil sample weight). This means that low-organic soils within the wording of CSN-EN-ISO-14688-2[73], or soils with a content of up to half the range defined for low-organic soils, are acceptable (comply with the criterion). Medium and high organic soils are generally not suitable for foundations..

In accordance with CSN EN ISO 14689-1[74], various types of peat[75], gyttja[76] and humus[77] are considered organic soils, i.e. material of Quaternary age. Organic matter is also a constituent of older geological rock formations, such as coal, caustobiolites and sediments containing the same.

Organic substances in soils reduce their permeability, enhance their compressibility, and they contribute to water and soil aggressiveness. The determination of the content of organic substances is one of the tests performed within soil classification for geotechnical purposes.

The fact of whether the impact of such unfavourable foundation conditions may be eliminated by technical measures (removal or replacement) (naturally only in cases when such soils appear in the foundation base) is crucial for the application of the criterion.

The following characteristic in relation to the foundation soil properties arise from the text in Section 2.6.2.3 and from the results of the conducted engineering geological surveys:

Insufficient Bearing Capacity of the Foundation Soil

The relevant criterion excludes lands where the bearing capacity of foundation soils is less than 0.2 MPa. Soils with this bearing capacity have the modulus of deformation equal to E_{def} = 8 - 10 MPa or lower.

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73 CSN EN ISO 14688-2 (721003) - Geotechnical Investigations and Testing - Identification and Classification of Soils - Part 2: Principles for a Classification

74 CSN EN ISO 14689-1 (721005) - Geotechnical Investigations and Testing - Identification and Classification of Soils - Part 1: Identification and Description.

75 Organogenic sediment formed by alteration of plant material almost or completely under water with lack of oxygen.

76 Dark-coloured, muddy deposit formed in oxygen-rich backwaters from dead vegetation cover, plankton and fine alluviums.

77 Humus is the product of alteration of organic substances (mainly dead plant tissue), i.e. it is the result of humification, which constitutes a complex chemical and biological process involving rotting, mouldering, decay and fermenting.

Laboratorily verified physical and descriptive properties of soils on the ETE3,4 construction site determine E_{def} = 10 MPa for Quaternary soils, E_{def} = 20 MPa for eluviums, E_{def} = 30 - 35 MPa for highly weathered rock, and E_{def} = 100 – 300 MPa for weathered rock (see Table 9 on page 40 of report [L. 144]). Thus, there are no foundation soils with bearing capacity of less than 0.2 MPa on the construction site.

Collapsibility

Investigations conducted on the construction site of Temelín NPP revealed no collapsible foundation soils (see basic materials [L. 144] to [L. 150]).

Swelling In the near region of Temelín NPP, no basic parent rocks are present and no Tertiary sediments nor products of Tertiary weathering have been preserved that could be classified as expansive soils. This means that soils capable of swelling are not found on the construction site of Temelín NPP (cf. basic materials [L. 144] to [L. 150] and [L. 165]).

Organic Content

The share of organic constituents in parent metamorphic rocks and their eluviums is excluded by their genesis. Only Quaternary cover may contains organic matter. However, as these will be removed during rough ground shaping, they do not form the foundation soils of constructed structures.

2.6.7.7.4 Criterion According to 5(b) (Decree No. 215/1997 Coll.)

The requirement consisting in the evaluation of "the unfavourable properties of the foundation soils, surrounding soils and rocks" may be generalized to the "unfavourable conditions for the foundation and construction of a nuclear installation". With the exception of the foundation soil properties specified by the criterions in Sections 4(k) and (g), the following factors may be included:

- Uneven rock weathering at the foundation bottom manifested by variable values of the E_{def} modulus of deformation and the penetration of highly weathered rock zones deep below the foundation bottom;
- Extremely high intensity of rock jointing (degree 6 or 5 according to CSN EN ISO 14689-1) and the penetration of such zones of jointed rocks deep below the foundation bottom;
- Changes in the physical and mechanical properties of soils and sedimentary rocks in consequence of saturation by water, mechanical load and vibrations;
- Unfavourable morphological conditions, e.g. extensively dissected and sloping terrain (over 15°) complicating earthwork on the construction site;
- High rock strength predetermining their inclusion into the 3rd workability class with respect to the prevailing volume of the presumed earthwork. This property should be namely evaluated in cases when the construction of a nuclear installation is to take place in the vicinity of another nuclear installation that is already in operation and when it is necessary to limit the use of explosives for rock loosening.

The territory is acceptable for the siting if rational technical measures may be implemented to attenuate or to eliminate the influence of these factors.

The following characteristic in relation to the foundation soil properties arise from the text in Section 2.6.2.3 and from the results of the conducted engineering-geological surveys:

Uneven Soil Weathering

Section 2.6.2.3 interprets the rock massif on the ETE3,4 construction site as an irregularly surface-weathered geological block. Nevertheless, with a view to the presumed foundation depth of the crucial structures of ETE3,4, this factor will not affect the quality of the structure foundations in a significant manner.

The upper part of the "zone of prevalent rock weathering" with values of the modulus of deformation E_{def} < 100 MPa has been largely removed from the area designated for the foundations of the crucial structures of ETE3,4 (areas S1 and S2) during rough ground shaping. The map in Fig. 24 (in Section 2.6.2.3) indicates that the zones where prevalent rock weathering reaches below 500 metres above sea level [78] are spatially limited. The rocks in these zones, if they appear in the Unit building foundation bottom, may be removed from the subsoil and replaced with plain concrete sealing. This procedure is technically feasible and it has been used during the construction of the NPP1,2 Units, as well as during the SFR construction in the site area of Temelín NPP.

Extreme Rock Jointing

The graphic drill logs (see basic material [L. 159]) suggests that zones with very close or extremely close discontinuity spacing are present in the rock mass on the construction site of Temelín NPP. The evaluation of the graphic drill logs of 34 drills of the JV[79] series implies that these zones are more frequent in inserts of granitic rocks[80] or in tectonic failured or altered metamorphites. The thickness of these zones is not more than 1 m in most drills. In contrast with metamorphic rocks, granitic rock zones with more intense jointing did not correlate too extensively with zones exhibiting lower values of the deformation modulus E_{def}. The deformation modulus of zones with very close to extremely close discontinuity spacing in metamorphic rocks[81] typically ranged between 15 and 30 MPa. A thicker zone of jointed, tectonic failured rocks with E_{def} equal to 30 or 15 MPa were revealed during drilling rather rarely. For example, drill JV 324 in the central part of area S1 found such paragneiss zone at 487 metres above sea level. The zone thickness was 10 metres. The value of the deformation modulus between 487 and 481 metres above sea level was equal to 30 MPa and 120 MPa at a greater depth (see basic material [L. 144]).

With regard to the rare occurrence of higher-thickness zones with extreme rock jointing, significant problems are not expected during the foundation of the crucial structures of ETE3,4. Moreover, this unfavourable property of the rock mass may be eliminated by a feasible technical measure, namely by the sealing of the foundation bottom.

[78] The actual foundation depth of the NPP3,4 MGU will probably be below 500 m a. s. l. The precise depth is unknown for the time being due to the competition for the construction contractor that is currently underway.

[79] Field tests were performed in these boreholes, e.g. presiometric measurements or logging and the documentation pertaining to the cores also contained an evaluation of the jointing of the rock mass.

[80] Rocks designated by the "C" letter code.

[81] Rocks designated by the "D" letter code.

Changes in the Physical and Mechanical Properties of Soils

With a view to the fact that the foundation materials of soil character are either not present at the foundation level or will be removed with the layers above the foundation level, the discussed changes in the soil properties are not expected to have any unfavourable effect on the quality of the foundations of the structures of ETE3,4. Based on the current findings on the ETE3,4 construction site, it is possible to presume that the crucial structures of ETE3,4 will be founded on rocks with the modulus of deformation E_{def} = 100 MPa and more that have sufficient bearing capacity and are not prone to plastic deformation or changes in their consistency (i.e. bearing capacity) due to saturation by groundwater.

Unfavourable Morphological Conditions

There are no unfavourable morphological conditions on the ETE3,4 construction site. The analysis of the terrain sloping (see Fig. 47) indicates that the construction site is situated on an artificially levelled plane divided into three elevation levels: 507 metres, 505.5 to 503.5 metres and 503 metres above sea level. Moreover, the decisive part of earthwork (rough ground shaping) has been performed within the process of preparation of the Temelín NPP main construction site.

High Rock Strength

The difficulties arising from this property should be considered with respect to the construction of ETE3,4 since it will collide with the operations of the NPP1,2 units.

According to the evaluation detailed in [L. 159], workability class 3 embraces slightly weathered and fresh paragneisses, pegmatites and vein granites of strength classes R2 and R1. Furthermore, the evaluation implies that the major part of the extracted soils will pertain to workability classes 1 and 2. Workability class 1 namely includes backfills in area S1 and completely weathered paragneisses in area S2. The prevailing presence of rocks belonging to workability classes 2 and 2, 3 is expected in deeper excavation pits (paragneisses and other rocks of strength classes R4 and R3). The occurrence of workability class 3 rocks is anticipated to be less frequent, however, with the exception of the area designated for the siting of the reactor buildings.

This unfavourable property of the rock massif may be overcome with the aid of a technical solution. Standard methods include, for example, the use of special hydraulic hammers for rock disintegration, rippers or the implementation of controlled excavation technologies. Also other methods exerting a limited seismic load on the surrounding environment may be applied, such as alternative non-explosive technologies.

2.6.7.7.5 Summarized Evaluation

The above indicated evaluation implies that the ETE3,4 construction site is not in conflict with the exclusion criterion defined in Section 4(k) of Decree No. 215/1997 Coll. [L. 1] and that the conditions laid down in the conditional criterion in Section 5(b) thereof have not been fulfilled. Moreover, the conducted supplementary survey [L. 165]) satisfied the requirements arising from Paragraphs 3.41 to 3.42 of the IAEA Safety Requirements No. NS-R-3 [L. 6].

2.6.7.8 CRITERION AND REQUIREMENTS ACCORDING TO SECTION 7.8

2.6.7.8.1 Specification of Risks Grouped in Section 7.8

Section 7.8 (Tab. 116) covers a part of the exclusion criterion specified in Section 4(g) of Decree No. 215/1997 Coll. [L. 1], which concerns the potential for liquefaction of soils on the construction site and threats to the stability of the basement of a nuclear installation. In addition, the section deals with the requirements specified in Paragraphs 3.38 to 3.40 of the IAEA Safety Requirements No. NS-R-3 [L. 6].

2.6.7.8.2 Criterion According to Section 4(g) (Decree No. 215/1997 Coll.) and Paragraphs 3.38 to 3.40 (NS-R-3)

The decrees and the IAEA Safety Guides and Safety Requirements give considerable attention to this issue and they consider the risk of soil liquefaction an exclusion criterion. Paragraph 3.40 of the IAEA Safety Requirements No. NS-R-3 [L. 6] stipulates: "If the potential for soil liquefaction is found to be unacceptable, the site shall be deemed unsuitable unless practicable engineering solutions are demonstrated to be available." Emphasis is laid on an in-depth investigation of the subsurface materials and on a determination of the potential for liquefaction of the subsoils. A detailed description of the process of evaluating the liquefaction potential is provided in Paragraphs 3.15 to 3.25 of the IAEA Safety Guide No. NS-G-3.6 [L. 13].

Soils susceptible to liquefaction are usually non-cohesive soils, such as fine or medium-grained sands with a small part of fine-grained soil located below the groundwater level. Another factor contributing to liquefaction is when soils are poorly graded (even-graded), intensely porous (40% and more), and completely water-saturated (saturation level Sr = 1). Events inducing liquefaction may embrace earthquakes, technical seismicity or exposure to groundwater flow pressure caused by construction interventions, etc. If such soils are found on the construction site, they should be characterised in detail and their depth and spatial extent should be determined. A construction site where soils susceptible to liquefaction cannot be removed from the subsoil of the crucial structures of a nuclear installation by means of a rational technical measure shall be deemed unsuitable.

During several investigation phases (1979-1989 and 2010), the ETE3,4 construction site was covered by a dense network of core holes with depths ranging from 10 to 50 m. Drilling was followed by laboratory tests of the rock and soil samples. The soils were subjected to a sieve analysis, based on which they were classified in compliance with the valid technical standards. Rocks were also classified according to their strength and weathering degree. The layout of the exploratory core holes is shown in Fig. 54.

The evaluation should namely consider the following:

- The original soils and weathered rocks of the crystalline complex were removed within the process of preparation of the main construction site of Temelín NPP (they were later replaced by backfill);
- The ETE3,4 structures will be founded on bedrock.

The evaluation therefore included rocks of strength classes R1 to R3 (pursuant to CSN EN ISO 14689-1)[82]. Even if these rocks may appear in the foundation bottom, it is more likely that they will be removed and replaced with a concrete sealing.

82 The basic materials from the investigations carried out between 1979 and 1989 use the IRSM scale (International Society for Rock Mechanics). Classes R0 to R2 correspond to this scale.

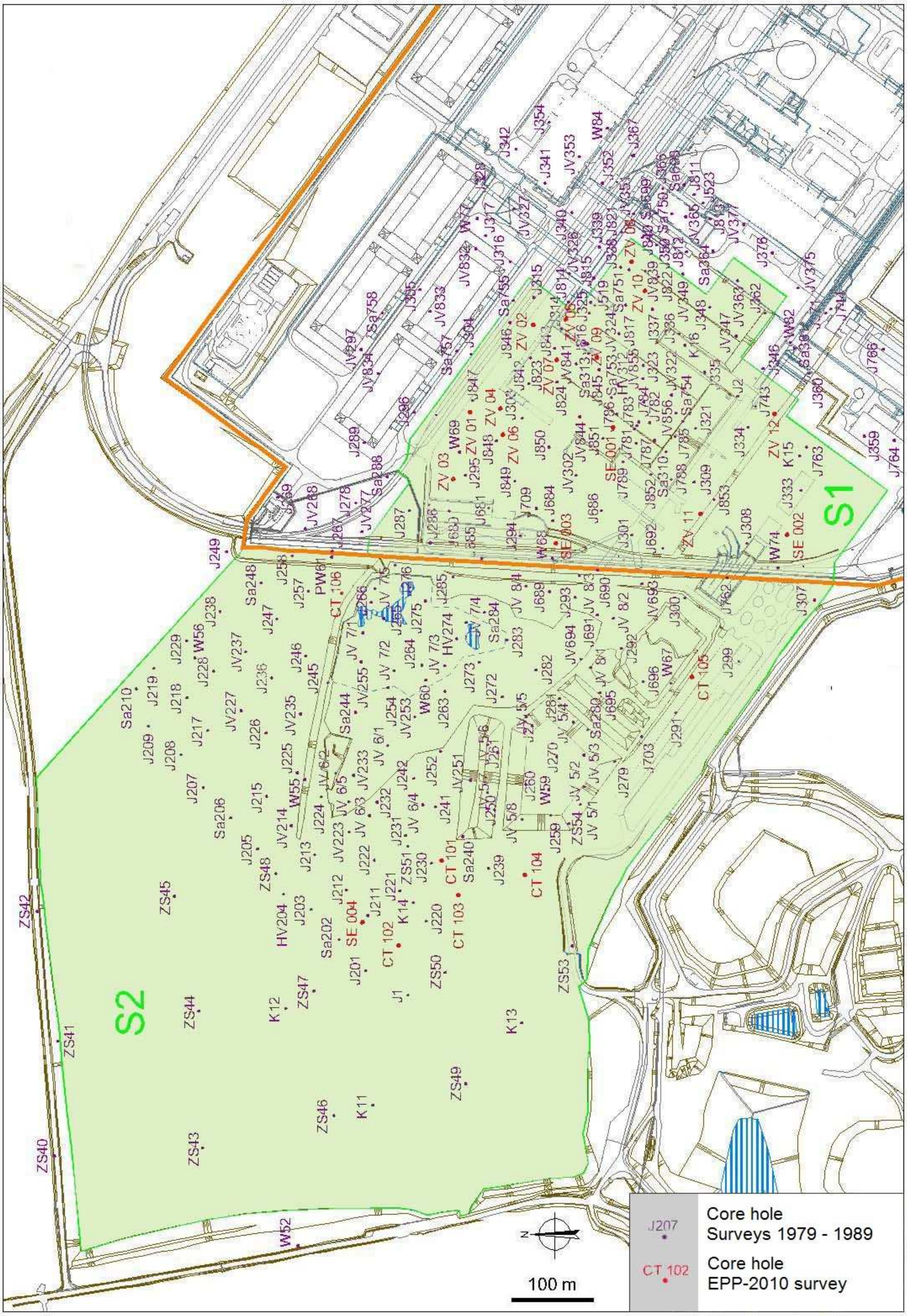

Fig. 54 Map of drilled engineering geological core holes in areas S1 and S2 during several phases of investigations carried out on the construction between 1979 and 1989 [L. 144] to [L. 146], [L. 182] to [L. 183] and 2010 [L. 165].

The physical and mechanical parameters that are relevant for the evaluation of the potential for liquefaction of the soils belonging to the above specified strength classes are available, for example, in [L. 145] (HS-2281-2-000.pdf - "Report on the Results of Supplementary Laboratory Test of the Rocks and Soils Found on the Site of Construction of the Reactors on the Main Construction Site of Temelín NPP"). Rocks samples (6 samples) of classes R1 to R3 from drills JV-251, JV-839, JV-841, JV-844 and JV-854 at elevations from 495.62 to 501.77 metres above sea level were collected at the ETE3,4 construction site and subjected to testing. The porosity (n) value of these samples - weathered paragneiss - ranged between 9.4 and 16.2% and the degree of saturation (S_r) ranged between 0.27 and 0.41.

As regards the grain size distribution of the weathered paragneiss of soil character, these were classified as gravelly-silty sands, clayey, silty and gravelly sands or sandy gravels (see basic material [L. 165]). The uniformity coefficient (c_u) was 46 and more, whereas the value should be > 15 for well-graded non-cohesive soils according to CSN EN ISO 14688-2. The coefficient of curvature (c_c) also corresponded to well-graded soil (the value ranged between $1 < c_c < 3$), while some samples were gap-graded ($c_c = 0,5$).

It is therefore possible to affirm that the completed investigations did not find any soils on the ETE3,4 construction site (areas S1 and S2), the properties of which would indicate their susceptibility to liquefaction.

This implies that the ETE3,4 construction site is not in conflict with the relevant exclusion criterion defined by Decree No. 215/1997 Coll. [L. 1] Concurrently, the requirements stipulated in Paragraphs 3.38 to 3.40 of the IAEA Safety Requirements No. NS-R-3 [L. 6] have been fulfilled.

2.6.7.9 CRITERIA AND REQUIREMENT ACCORDING TO SECTION 7.9

2.6.7.9.1 Specification of Hazards Grouped in Section 7.9

Section 7.9 (see Tab. 116) groups together the geological phenomena that represent hazards during the construction of the of a nuclear installation in an underground cavity[83]. Decree No. 215/1997 Coll. [L. 1] incorporates these hazards in the criteria defined by Section 4(l), Section 4(m), Section 5(g), and Section 5(h).

The IAEA Safety Guides, which concern the evaluation of nuclear installations having the character of a nuclear power plant, do not contain any specific requirements in relation to the evaluation of nuclear installation of this type that is sited underground.

[83] The underground structure mentioned in the text of the criterion in Section 4(m) is understood as a structure executed in an underground cavity created by driving, when the entire construction takes place underground without any interventions in the overlying rock. Decree No. 61/1988 Coll., on Mining Activities, Explosives and the State Mining Administration, as amended, introduces the term "underground work" to designate the underground cavity created by driving through its amendment No. 376/2007 Coll. For details go to the website of the State Mining Administration (http://www.cbusbs.cz/).

2.6.7.9.2 Summarized Evaluation

The indicated criteria do not apply to the evaluation of ETE3,4.

2.6.7.10 CRITERIA AND REQUIREMENT ACCORDING TO SECTION 7.10

2.6.7.10.1 Specification of Hazards Grouped in Section 7.10

Section 7.10 groups together the hazards that are incorporated in the criteria defined in Section 4(j) of Decree No. 215/1997 Coll. [L. 1] and Section 5(d) of Decree No. 215/1997 Coll. [L. 1] (see Tab. 117). The table also includes the requirements stipulated in Paragraph 4.7 of the IAEA Safety Requirements No. NS-R-3 [L. 6] that pertain to the evaluated phenomena.

2.6.7.10.2 Criterion According to Section 4(j) (Decree No. 215/1997 Coll.)

The purpose of the criterion is to ensure that groundwaters are protected from deterioration by radioactive substances, the origin of which is associated with the operated nuclear installation.

The first step of the evaluation process is to determine whether the hydrogeological zone falling within the site vicinity is a "significant resource of groundwater". When assessing the significance of the groundwater resource, a helpful aid may be the classification of rocks based on their transmissivity [L. 123]). In this document, rocks are divided into six classes (see Tab. 129), whereas rocks of class I and II (very high and high aquifer transmissivity) have the highest potential for becoming a significant resource of groundwater, as well as class III rocks with higher transmissivity levels (medium aquifer transmissivity).

Tab. 129 Classification of rocks based on their transmissivity and approximate conversion of the filtration parameters expressing transmissivity (modified according to [L. 123]).

Class	Aquifer transmissivity designation (rock complex)	Transmissivity coefficient T [m^2/day]	Specific yield q [$ls^{-1}m^{-1}$]	Water-management significance The transmissivity value indicates an environment with the following groundwater utilisation preconditions
I	Very high	>500	>5.0	Large concentrated withdrawals
II	High	100-500	1.0-5.0	Concentrated withdrawals of lesser significance
III	Medium	10-100	0.1-1.0	Dispersed, mostly minor withdrawals for local supply
IVa	Low	1-10	0.01-0.1	Individual, irregular withdrawals for local supply
IVb	Very low	0.1-1	0.001-0.01	
V	Negligible	<0.1	<0.001	Securing of individual sources for local supply is mostly difficult and often impossible even in case of very limited consumption

Where the site vicinity falls within areas with transmissivity class I and II aquiferous soils, it is necessary to consider whether the interest in protecting groundwaters should not take priority over the interest in siting the nuclear installation in such area (site vicinity). In such cases, the considerations may be supported by the rationalization provided in the Directive 2006/118/EC of the European Parliament and of the Council on the protection of groundwater against pollution and deterioration**.**

As regards mineral waters, it is in the public interest to protect these resources, especially due to their unique properties when applied for balneological purposes. The protection of such resources is ensured by delimiting protective zones of natural healing sources and sources of mineral water. These zones are established by a decree of the Ministry of Health. Accordingly, another decree of the Ministry of Health protects sources of natural mineral water that are used for the production of bottled mineral water.

Conceivable conflicts of interest (conflict with interests involving the protection of groundwater or mineral water resources) should be resolved outside the siting process (i.e. when looking for a suitable construction sites), namely within the spatial planning process.

The first step of the process of evaluation of the site vicinity of ETE3,4 focused on determining whether the condition of the criterion was fulfilled, i.e. whether the site vicinity falls within an area, which is a "significant resource of groundwater or mineral water". Three criteria were specifically devised for evaluation purposes, in particular:

- Whether the site vicinity falls within a hydrogeological zone[84], which is considered significant from the point of view of groundwater reserves, which bears recoverable reserves of groundwater and is utilised for mass-supply water withdrawals;
- Whether the site vicinity falls within any of the "Protected Areas of Natural Water Accumulation" (CHOPAV);
- Whether areas with sources of mineral water may be found in the site vicinity, whether the site vicinity collides with the protective zones of natural healing sources and sources of mineral water or with areas where springs of healing balneological mineral waters are located.

Evaluation of the Collision of the Site Vicinity with a Significant Hydrogeological Zone

In the explanatory notes to the geological maps [L. 195],[L. 134]) and in other basic materials (e.g. [L. 51], [L. 113]), the site vicinity of ETE3,4 is rated as a less significant hydrogeological structure.

The lesser significance of the area of the crystalline complex also reflects itself in the very limited extent of hydrogeological exploratory drilling, in the small number of hydrogeological boreholes and wells or larger groundwater intake structures. Nonetheless, no large-scale hydrogeological survey focused on assessing the recoverable reserves of groundwater is planned in the site vicinity area[85] (cf. also Fig. 55).

84 A hydrogeological zone is a territory with similar hydrogeological conditions, saturation types and groundwater circulation. The indicated definition is introduced in Act No. 254/2001 Coll. The list of hydrogeological zones is provided in Decree No. 5/2011 Coll Data on the borders and locations of hydrogeological zones are recorded in compliance with Section 22(4)(a)
(Act No. 254/2001 Coll. [L. 284].

85 At present, the Czech Geological Survey has launched the "Reassessment of groundwater reserves" project, the first phase of which should asses 56 selected units – i.e. hydrogeological regions. The NPP3,4 site vicinity area was not included in the project. For details see http://www.geology.cz/rebilance/rajony.

Crystalline rocks form a rock complex of very low permeability with relatively higher permeability of the weathering mantle in the subsurface fissure disintegration zone, in tectonic failure zones, and in more rigid rock inserts. In these zones, a network of open fissures is created, which allows the flowing of groundwater and the formation of small-sized groundwater aquifers.

We may thus conclude that the site vicinity of ETE3,4 does not fall within a hydrogeological region that is considered significant from the point of view of groundwater resources.

Evaluation of the Collision of the Site Vicinity with the Protected Area of Natural Water Accumulation

The map in Fig. 55 clearly shows that the site vicinity of ETE3,4 does not collide with any protected area of natural water accumulation.[86,87,88]

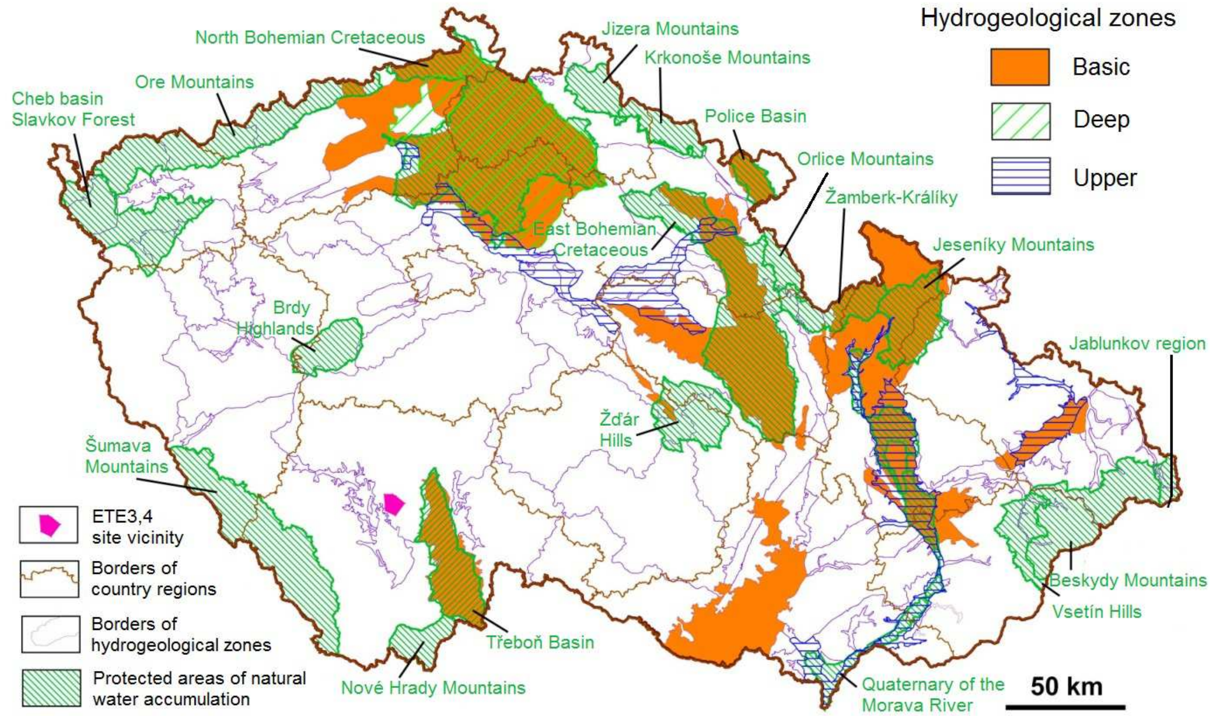

Fig. 55 Map of the hydrogeological zones of the Czech Republic, the regions included in the "Reassessment of groundwater reserves" project [L. 172], which are divided into basic, deep and upper zones (taken from http://www.geology.cz/rebilance/rajony), and protected areas of natural water accumulation (without differentiating surface or groundwater accumulation).

[86] See Order of the Government of the Czech Socialist Republic No. 40/1978 Coll. on the protection of areas of natural accumulation of water in the Beskydy, Jeseníky, Jizera, Krkonoše, Orlice, Šumava Mountains and the Žďár Hills.

[87] See Order of the Government of the Czech Socialist Republic No. 40/1978 Coll. on the protection of areas of natural accumulation of water in the Brdy Highlands, Jablunkov region, Ore Mountains, Nové Hrady Mountains, Vsetín Hills and Žamberk-Králíky.

[88] See Order of the Government of the Czech Socialist Republic No. 85/1981 Coll. on the protection of areas of natural accumulation of water in the Cheb Basin, Slavkov Forest, North Bohemian Cretaceous, East Bohemian Cretaceous, Police Basin, Třeboň Basin and the Quaternary of the Morava River.

The area closest to ETE3,4 is the "Třeboň Basin", namely its northern part. The erosion base of the Vltava River separates it from Temelín NPP. It is unlikely to presume any effects on the mentioned area. They are practically ruled out by the morphology of the terrain, the character of the rock environment and by the gradient conditions (see basic material [L. 51]).

Evaluation of the Collision of the Site Vicinity with Sources of Mineral Water

As the Map of Mineral Waters indicates (see basic material [L. 84]), the site vicinity of ETE3,4 does not collide with any plotted sources of mineral water.

Considering the result of the first step of the evaluation process, which suggests that the site vicinity does not collide with any of the above specified hydrogeological indicators, a more detailed investigation of the relevant hydrogeological structure has not been performed (cf. proposed evaluation procedure in basic material [L. 166]).

2.6.7.10.3 Requirement According to Paragraph 4.7 (NS-R-3)

In connection with the evaluation of the hydrogeological hazards, Paragraph 4.7 of the IAEA Safety Requirements No. NS-R-3 [L. 6] specifies the requirement to describe the hydrogeological conditions in the relevant area, to characterise the main water-bearing formations, their interactions with surface waters, and to collect data on the use of groundwater in the area under review.

Characteristics of the Main Water-Bearing Formations in the Site Vicinity

According to hydrogeological regionalization, the site vicinity belongs to Region 6320 – the Middle Vltava Crystalline Complex, the southern part of the site vicinity partially extends to Region 2160 – the České Budějovice Basin. The site vicinity of Temelín NPP is plotted on four pages of the 1 : 50,000 scale hydrogeological (see basic materials [L. 65], [L. 116], [L. 124], [L. 98]).

These basic materials indicate that two basic types of hydrogeological map environments may be found in the site vicinity area of Temelín NPP:

- The crystalline complex;
- Sediments of cover formations represented by Tertiary sediments and Quaternary fluvial deposits of the Vltava River.

The map [L. 65] shows Quaternary fluvial deposits of the Vltava just outside the southern border of the site vicinity area. With transmissivity ranging from 3.10^{-6} to 10^{-3} m^2s^{-1}, they represent the most significant aquifer in the area under review.

Neogene sediments may be found in the form of several relics in the southwestern (near Dříteň), western (between Lhota pod Horami and Malešice), and the northern (near Bohunice) part of the site vicinity. They are distinguished by extensive facial diversity. Groundwater is only bound to incoherent sand positions and inserts with intrinsic permeability (see lit. [L. 177]). The transmissivity of Tertiary deposits ranges from 10^{-6} to 10^{-5} $m^2.s^{-1}$ (see basic materials [L. 65], [L. 116], [L. 124], [L. 98]).

Two mutually independent and spatially separated groundwater bodies (horizons) may be found in the crystalline complex:

- The shallow groundwater horizon bound to Quaternary sediments and the near-surface eluvium zone, and the near-surface fissure disintegration zone;
- The fracture groundwater horizon (designated as "main" in lit. [L. 51]) bound to the fracture system of the deeper bedrock.

The shallow groundwater horizon is incoherent with low permeability, occasionally slightly under pressure and somewhere with strongly fluctuating groundwater table due to climactic effects. The horizon is characterised by intrinsic-fissure permeability, which changes to fissure permeability with increasing depth. Even though paragneisses contain a dense joint network, but joints are mostly closed and contain only adhesive water. The weathering of paragneisses results in the formation of eluviums, which are a parent material of clayey, poorly permeable soils.

The main aquifer is located at a depth of 50 metres and more below the surface of the territory. The saturated zone is bound to the crystalline rock joint system. Based on the findings revealed by very deep boreholes (see e.g. lit. [L. 151]), the rock mass manifests very low fracture permeability up to the depth of 150 to 200 metres, and almost no permeability at greater depths, with the exception of occasional tectonic failures (NW–SE, NE–SW, N–S direction). Drill Js 799, which was drilled (and backfilled) to the depth of 731 metres in the site area of Temelín NPP (lit. [L. 151]), detected slight water bearing at depths: 435 m, 543 m, 572 – 577 m below the terrain (see lit. [L. 51]). The yield reaches up to 10^{-1} $l.s^{-1}$.

The water of the main aquifer may be characterised as stagnating or very slowly flowing groundwater. It is of Holocene age and under natural conditions, it has no direct contact with the surface (see lit. [L. 184]).

Slightly higher fracture permeability of the rock massif may be presumed in areas formed by migmatitised rocks of the crystalline complex. In the site vicinity, it is namely the area to the north and northwest of the Vodňany Mylonite Zone (SW–NE direction). Mostly biotite gneissose granites and leucocratic migmatites of the Podolí Complex may be found in the area. These rocks are fragile, yet the joint network is much less dense. A major part of the fractures is open only in the subsurface zone where shallow reservoirs of plain groundwater are formed, with

considerably fluctuating water levels depending on the character of the fractures. Weathering alters these rocks into sandy eluviums with good intrinsic permeability (see lit. [L. 177]).

Refoliated paragneisses may be found in the area of the crystalline complex of Týn nad Vltavou south of the Vodňany Mylonite Zone. In this part of the crystalline complex, areas with slightly higher fracture permeability are spatially limited merely to inserts of rigid, densely fractured rocks (aplite, pegmatite, vein quartz), usually only a few metres thick. More intense water bearing may be expected near discontinuities. Geological maps (see basic materials [L. 194], [L. 135]) do not display any verified faults in the site vicinity (cf. Fig. 42). Assumed faults were delimited only on the basis of geophysical (magnetic) indications (see basic materials [L. 195], [L. 134]).

Groundwater and Surface Water Interactions in the Site Vicinity

The upper aquifer in the site vicinity is fed by infiltration of atmospheric water. As mentioned above, it does not communicate with the deeper (main) aquifer (see lit. [L. 184]).

Communication of the upper groundwater aquifer with surface water is effected by means of valley springs and wetlands in terrain depressions, as well as dense melioration networks in cultivated areas.

Surface water from the eastern part of the site vicinity is drained to the Vltava and from the western part to the White Creek [Bílý potok], which belongs to the Blanice River catchment area. These water courses represent the local erosion bases of the site vicinity area. Furthermore, the site vicinity may be divided into eight partial catchment areas of the fourth order, four of which are linked to the area of the construction site of Temelín NPP. These are the catchment areas of the Paleček Creek [Palečkův potok] 1-06-03-077 and the Strouha Creek 1-06-03-073, which mouth into the Vltava River, and the catchment area of the Malešice Creek [Malešický Brook] 1-08-03-079 and the Temelínec Creek [Temelínecký potok] 1-08-03-079/2 belonging to the Blanice catchment area.

A more detailed evaluation of the interaction of groundwater with surface waters on the construction site of ETE3,4 is provided in Section 2.6.7.11.2.

Data on the Use of Groundwater in the Site Vicinity

The municipalities in the surroundings of Temelín NPP are connected to public water mains (Malešice, Sedlec, Kočín and Temelín). The municipality of Temelín, which is closest to areas S1 and S2, lies above the area of infiltration of Temelín NPP. The remaining municipalities in the close surroundings of the Temelín NPP construction site were cleared of all inhabitants in the 1980s (Temelínec, Křetěnov, Podhájí, Březí near Týn nad Vltavou, Knín). At present, there are only two secluded settlements with nine reported inhabitants near the stronghold of Býšov that belong to the municipality of Knín. The settlements are not connected to the public water mains and water is supplied from local wells. The wells are located outside the area potentially affected by the construction and the operations of ETE3,4 (for details see [L. 179]).

The nearest more significant withdrawal point is located approximately 1.5 km northeast of the site vicinity in the surroundings of Čihovice[89] not far from Týn nad Vltavou. Another withdrawal point is marked in the database of the Hydroecological Information System of the T. G. Masaryk Water Research Institute (http://heis.vuv.cz/). The withdrawn water supplies the operations of the Wienerberger brickworks. Both withdrawal points are separated from the area of the upper groundwater body potentially affected by the operations of Temelín NPP by the drainage base of the Paleček Creek [Palečkův potok].

The database mentioned in footnote 89 implies that no water withdrawal for public supply is planned in the site vicinity (see [L. 179]).

[89] The plan for development of the water supply and sewerage system on the territory of the South Bohemian Region. Available online at: http://www.kraj-jihocesky.cz/index.php

2.6.7.10.4 Summarized Evaluation

The above specified findings clearly indicate that site vicinity does not constitute an area with used or significant groundwater reserves, the quality of which could be jeopardized by the operations of ETE3,4.

Thus, the site vicinity of ETE3,4 is not in conflict with the exclusion criterion defined in Section 4(j) of Decree No. 215/1997 Coll. [L. 1]. It is also possible to confirm that the information and data required by Paragraph 4.7 of the IAEA Safety Requirements No. NS-R-3 [L. 6] were collected.

2.6.7.11 CRITERIA AND REQUIREMENT ACCORDING TO SECTION 7.11

2.6.7.11.1 Specification of Hazards Grouped in Section 7.11

Section 7.11 (see Tab. 117) groups together the criteria stipulated in Section 5(d) of Decree No. 215/1997 Coll. [L. 1] and in Section(f) of Decree No. 215/1997 Coll. [L. 1], which deal with the hydrogeological conditions on the construction site of a nuclear installation.

The evaluation performed in compliance with this section also incorporates the requirements defined in Paragraphs 3.43, 4.8 and 4.9 of the IAEA Safety Requirements No. NS-R-3 [L. 6].

2.6.7.11.2 Criterion According to Section 5(d) (Decree No. 215/1997 Coll.) and Paragraph 3.43 (NS-R-3) - Part 1

The criterion applies to hydrogeological structures with complicated groundwater occurrence and behaviour. The fact that it is difficult to monitor and predict groundwater behaviour substantially affects the evaluation of the environmental acceptability of a NPP structure, particularly with respect to the jeopardy of the potential uncontrolled spreading of radioactive substances.

Typical examples of unsuitable properties include environments with alternation of permeable and non-permeable layers, multiple groundwater bodies with different buyoancy height, structures with irregular changes in the watershed divide, environments with rapidly changing fracture permeability, etc. (see basic material [L. 186]).

The phenomena described by the criterion may also occur in cases when a formation with karstic permeability is a part of the hydrogeological structure or if the hydrogeological structure becomes disrupted due to the withdrawal of liquids or the mining of mineral resources (see basic material [L. 166]).

The requirement to study the "groundwater regime" is also defined by Paragraph 3.43 of the IAEA Safety Requirements No. NS-R-3 [L. 6]. The constructional hydrogeological properties of the site are mentioned in the requirements, which specify the scope of the factors to be investigated (considered) in the site verification stage (see Paragraph 2.8 of the IAEA Safety Guide No. NS-G-3.6 [L. 13]). One of the points specifically states "groundwater levels and regimes". A detailed analysis of the groundwater regime and its physico-chemical properties is included in the recommendation concerning the site verification stage (see Paragraph 2.19 of the IAEA Safety Guide No. NS-G-3.6 [L. 13]).

The resolution of this criterion and requirement is primarily based on the results of several years of monitoring of various groundwater parameters within the framework of the project entitled "Monitoring and Evaluation of the Quality of Surface and Groundwaters and Their Change in Connection with the Impact of the Construction and Operation of the Temelín Nuclear Power Plant on Its Surroundings" (see lit. [L. 89]).

From a general perspective, the groundwater regime on the construction site of Temelín NPP and its close surroundings may be characterised as follows:

- The natural groundwater regime, the sequence of water-bearing horizons and their hydraulic properties were notably changed by the preparation of the construction site of Temelín NPP and the subsequent construction of NPP1,2;
- The groundwater draining conditions on the construction site are relatively stabilised at present;
- There have been no substantial changes in the direction of the flow of the groundwater, which is drained from the site of the power plant into all directions with a relatively high level gradient.

It is also possible to say that a rock environment with unsuitable properties, which could impede groundwater monitoring within the meaning of the criterion stipulated in Section 5(d) of Decree No. 215/1997 Coll. [L. 1] and its interpretation (see basic materials [L. 186] and [L. 166]) is neither present on the construction site of Temelín NPP nor has it been created by the construction of NPP1,2.

Brief Characteristics of the "Water Quality Monitoring and Evaluation" Programme at Temelín NPP

A monitoring system of hydrogeological observation, wells, and other sampling points has been established in the site area of Temelín NPP and its surroundings, which serves for regular evaluations of groundwater level fluctuations on the construction site, for groundwater quality monitoring, groundwater (radio) activity monitoring, and for evaluations of the impact of the operations of the nuclear power plant and its facilities on groundwaters. Layout of the boreholes and wells is arranged in a manner allowing the evaluation of the monitored parameters with respect to the following structures and facilities:

- "Březí" other waste disposal facility in the locality of the former municipality of Březí;
- "Knín" other waste disposal facility in the urban area of the former municipality of Knín;
- Waste disposal facility in Temelínec designated for the storage of other waste, municipal waste and chemical sludge from the cooling water treatment plant;
- Oil and oil product storage facility - i.e. above-ground storage of fuel oil and turbine oil and underground storage of diesel oil and petrol;
- Site area of Temelín NPP - i.e. boreholes RK 1 to 8 near the Nuclear auxiliary service building, borehole RK 20 near the Essential water cooling spray basins, borehole RK 21 near the Chemical storage, borehole RK 22 below the Diesel generator station of Unit 1, boreholes RK 23, 24 and 25 behind the Turbine hall of Unit 1, borehole RK 26 near Spent fuel storage, and borehole HV 615 near the Cooling towers.

In addition, the monitoring system includes 12 objects (boreholes, wells) in the surroundings of Temelín NPP, which serve for the purpose of evaluations of the potential impact of the operations of Temelín NPP on groundwaters in the area.

The scope and the frequency of the monitoring of individual parameters is set out by the monitoring plan (for details see basic material [L. 89] and [L. 113]). The data from the period from 2000 to 2010 along with the results of control measurements (see basic material [L. 179]) were utilised within the evaluation of the construction site in compliance with the criterion defined in Section 5(d) of Decree No. 215/1997 Coll. [L. 1].

Water-Bearing Horizons in the Site Area of Temelín NPP And Its Close Surroundings

Two mutually independent and spatially separated groundwater horizons may be found in the intact rock environment on the site of Temelín NPP and its surroundings:

- Shallow groundwater horizon bound to Quaternary sediments and the near-surface eluvium zone, mostly along the boundary of Quaternary sediments and eluviums, or on the eluvium base and in the subsurface fissure disintegration zone;
- Fracture groundwater horizon bound to the fracture system of the deeper bedrock.

In areas disturbed by the preparation of the construction site of Temelín NPP and the construction of NPP1,2, a shallow discontinuous groundwater horizon is being formed and it is bound to more permeable positions of landfills, fill of various underground structures or lines, and backfills of excavation pits.

Shallow groundwater horizons, whether of natural or anthropogenic origin, are particularly important for evaluations according to this criterion.

As indicated by investigations (see lit. [L. 165]), a natural shallow groundwater horizon may be found in areas S1 and S2 in the subsurface fissure disintegration zone. It is bound to highly or slightly weathered rocks of the bedrock. Moderately higher water bearing structures are associated with bodies of rigid, intensely jointed rocks (aplites, pegmatites and vein quartz). Drill SE 003 in area S1 (see lit. [L. 165]), crossed a moist rock only at the depth of 10.9 metres in connection with a weathered pegmatite lense. The situation in area S2 was similar: groundwater levels were encountered on eluvium bases of the underlying rocks (CT-104), in the fissures of paragneisses (SE 004, CT-103), or near jointed pegmatite bodies (CT-106). See [L. 165]).

Shallow groundwater horizons of anthropogenic origin appear in areas S1 and S2 rather irregularly. At a higher elevation plane of S1 area (cf. Fig. 47), the drills drilled for the purpose of investigating the thickness of landfills in the area (drills ZV, see lit. [L. 165]) did not encounter groundwater at all. In area S1, groundwater table was encountered only in drill CT-101, namely near the landfill base.

Groundwaters of this horizon appear more frequent in excavation pit backfills, e.g. in S1 area in the excavation pits of the cooling water pumping stations of units 3 and 4 of the VVER 1000. These pits were caught by drills ZV-11 and ZV-12 (see lit. [L. 165]). Drill ZV-11 encountered groundwater table at the landfill bed of silty- sandy character (at 5.5 m below the terrain) and drill ZV-12 at the depth of 8.4 m below the terrain.

The saturation of backfills near the underground structures of buildings largely appears in the part of Temelín NPP that is currently in operation. This constructional hydrogeological problem is resolved by means of a drainage system (pumping wells). Forty-nine wells were built between the years 1994 and 1996. Withdrawals maintain the groundwater table within the range of 20.5 to 22.5 m below the terrain (481 – 488 m a. s. l.).

Groundwater-table contours and the Directions of Groundwater Flows

The shallow aquifer with rather intense water circulation is directly supplied by precipitation infiltration. Under natural conditions, the groundwater body is drained into springs and wetlands in terrain depressions (see also [L. 51]).

At present, the flow system is affected by agricultural melioration, landscaping in the site area of Temelín NPP and its surroundings (leading of upper courses into artificial underground waterways, building of discharge ditches, etc. - see below), as well as by withdrawal of groundwater from drainage wells.

The hydroisohypses showing the flow of groundwater along the surface horizon were developed with the aid of measurements carried out in the observation boreholes and wells of the Temelín NPP monitoring system, and based on findings on shallow groundwater effluents detected during the mapping Fig. 56.

The territory covered by the model is confined to the sector delimited by the catchment areas of the fourth order that are directly linked to the site area of Temelín NPP and it is concurrently delimited by the drainage points (forming the natural border of the area under review). Due to insufficient input data, the model does not include the catchment area of the Strouha Creek. Since it is in no relation with areas S1 and S2, the interpretation of the direction of flow in this catchment area is not crucial for this evaluation.

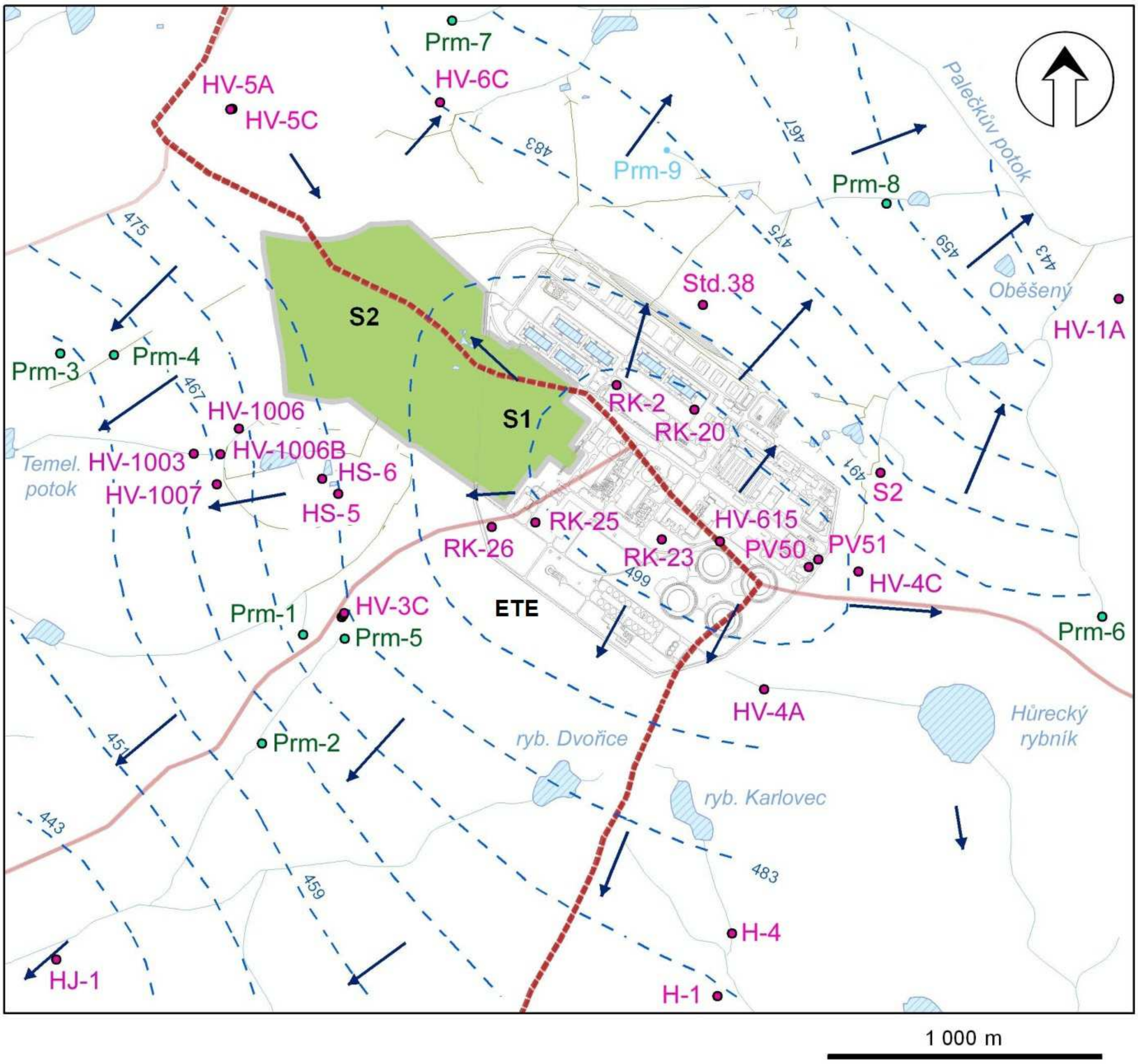

Fig. 56 Map of the hydroisohypses and the direction of the flow of groundwater in the upper horizon. Taken from [L. 179].

The plotted Groundwater-table contours clearly indicate that areas S1 and S2 are located in an infiltration area, the northern and eastern part of which is drained by the Paleček Creek [Palečkův potok] and the Strouha Creek into the Vltava River. Its southwestern and western part are drained by the Malešice Creek [Malešický potok] and Temelínec Creek [Temelínecký potok] to the Blanice River.

The ascertained flow directions indicated by the arrows confirm the above presented presumption, i.e. that the groundwater flows from the site area of Temelín NPP into all directions.

Groundwater and Surface Water Interactions in the Site Area of Temelín NPP and Its Close Surroundings

Natural interactions between the shallow aquifer and surface waters in the form of springs and wetlands in terrain depressions occur rarely in the site area of Temelín NPP and its close surroundings. Based on the results of terrain reconnaissance (see basic material [L. 179]), only Prm 6 spring is considered natural, while the influence of melioration cannot be ruled out in other cases. Prm 3 and 4 springs are obviously manifestations of interrupted melioration systems Fig. 56. It is also necessary to mention the extensive wetland area in area S1, which was formed on the site of the excavation pit of the cooling towers of the envisaged VVER 1000 units 3 and 4.

The predominant interaction method of ground and surface waters is the drainage of groundwater by means of the former, partially renovated storm sewer system in areas of demolished structures that were a part of the constructional plant or through melioration. The evaluation of this method of groundwater drainage is apparent from Fig. 57.

An analysis of the map materials and the results of the terrain reconnaissance imply that groundwater is drained from areas S1 and S2 and their vicinity in six directions.

Direction ➊ represents drainage of the western segment of area S2. Drainage is executed by means of the former, partially renovated storm sewerage, as well as through the backfills of these conduits. The entire sewer network is directed to the site of the waste disposal facility in Temelínec, from where water may be drained into the Temelínec Creek [Temelínecký potok].Direction ➋ designates drainage of the northern or northwestern segment of area S2. The method of groundwater drainage is similar to direction 1 and the recipient is the Paleček Creek [Palečkův potok].

Direction ➌ represents the system of the storm sewerage of Temelín NPP, which is in operation and diverts precipitation water from areas S1 and S2 and from the area of the Prestressed cable station. Atmospheric water is diverted by a collector into safety basins in Býšov and onwards into the drainage channel of the Strouha Creek. Groundwater may flow through the backfills of collector sewers and movement in the direction of its natural flow may be anticipated as well.

Direction ➍ designates drainage of the southern foreground of areas S1 and S2. The melioration system plays a fundamental role in the drainage of groundwater from these areas. The system mouths into a collector from where water is diverted into the Temelínec Creek [Temelínecký potok].

Direction ➎ represents drainage of the northeastern foreground of areas S1 and S2. Drainage is executed by means of the former, partially renovated storm sewerage, as well as by natural drainage in the direction of the sloping terrain. Two collectors mouth into the Paleček Creek [Palečkův potok] via settling tanks.

Area ❻ designates wetland on the site of the planned construction of the cooling towers of VVER1000 units 3 and 4. The wetland fulfils a drainage function as it draws groundwater from its surroundings.

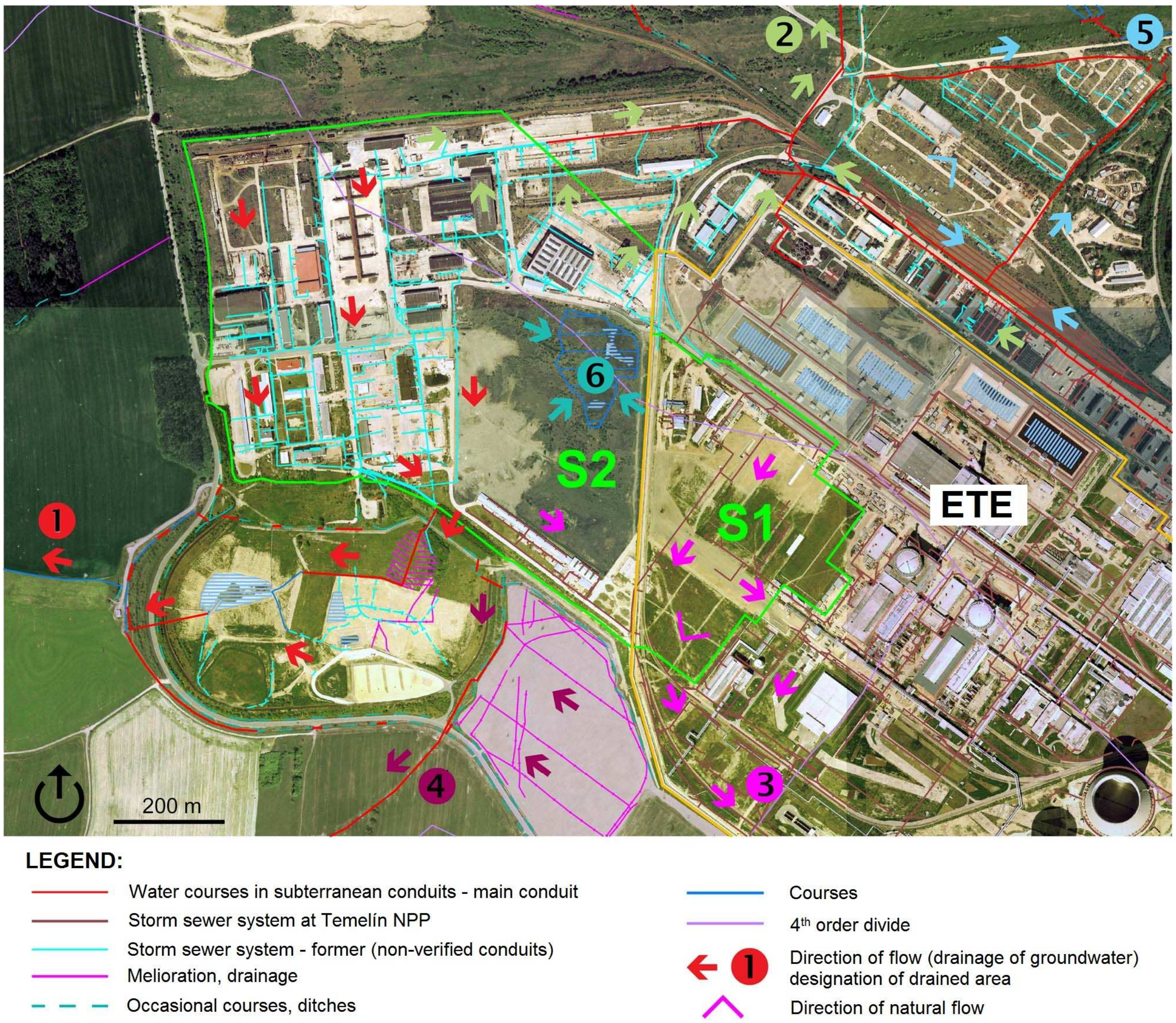

Fig. 57 Evaluation of drainage of groundwater from areas S1 and S2 by means of drainage elements. The layout of the conduits is plotted in an orthophotomap of Temelín NPP from 1995. The current state in areas S1 and S2 is shown in the orthophotomap from 2007 (see Fig. 25). The layout of the different types of sewerage was taken from the General_layout_plan_ETE_34_rev002.dwg (ČEZ, a. s., Temelín Nuclear Power Plant, 2012).

Possibility to Monitor the Behaviour of Groundwater in the Site Area of Temelín NPP and Its Close Surroundings

As mentioned in the introduction of this chapter, a number of parameters of the upper groundwater horizon in the relevant territory have been monitored in the long term and the measurement results are regularly evaluated.

The most recent analysis of the hydrogeological and drainage conditions in areas S1, S2 and their close surroundings (see basic material [L. 179]) identified the main directions of the flow of groundwater from the upper horizon. Both the natural flow and the flow affected by anthropogenic interventions into the hydrogeological structure were described.

Although the precipitation-runoff conditions are rather complicated on the construction site as a result of numerous interventions in the natural flow of groundwater, a reliable model of the runoff and the direction of the flow of groundwater in the upper horizon may be created. The development of this model was possible namely due to the existence of an extensive set of findings from monitoring boreholes along with records on the quantity of groundwater withdrawn for the purpose of lowering water levels, as well as detailed technical documentation of the power plant structures and facilities Fig. 57.

Further increase of the knowledge of groundwater behaviour will be incorporated in the subsequent phases of the execution of the ETE3,4 construction project, which will include the design and building of a system for the monitoring of ground and surface waters (respectively the extension of the existing monitoring system).

2.6.7.11.3 Requirements According to Paragraphs 4.8 and 4.9 (NS-R-3)

Paragraph 4.8 of the IAEA Safety Requirements No. NS-R-3 [L. 6] lays down the obligation to investigate the migration and retention characteristics of soils, the dispersion characteristics of aquifers, as well as the physical and physico-chemical properties of underground rocks and soils. The purpose of these investigations is to identify the transfer mechanisms and preferential pathways through which radionuclides may propagate by means of groundwater. Paragraph 4.9 of the IAEA Safety Requirements No. NS-R-3 [L. 6]) imposes the obligation to develop a suitable model facilitating an assessment of the potential impact of the contamination of groundwater on the population.

Monitoring of the potential propagation of radionuclides in a water environment is also mentioned in the general requirement defined in Paragraph 2.23, i.e. "the direct and indirect pathways by which radioactive material released from the nuclear installation could potentially reach and affect people and the environment shall be identified".

When dealing with the requirements specified in Paragraphs 4.8 and 4.9 of the IAEA Safety Requirements No. NS-R-3 [L. 6], we start from the basic materials processed during the execution of the current NPP1,2. Research focusing on the hydrogeological characteristics of the rock environment will be carried out within an in-depth hydrogeological survey of the ETE3,4 construction site.

The following text provides an overview of the available migration characteristics of the environment[90] that may be determined by hydrogeological investigations. The specific values of individual parameters ascertained by experiments or field tests are not presented as they are valid only for the investigated portion of the rock environment (i.e. the vicinity of boreholes RK-2, RK-4 and RK-5). Locally valid values will have to be determined for the ETE3,4 construction that will apply to the rock environment partially created after the completion of the construction, e.g. within the extension of the system of radiation control boreholes.

The aim of the following subsection is to demonstrate that groundwater behaviour in relation to the potential propagation of radionuclides may be monitored on the

[90] The migration characteristics of the environment are generally defined as parameters characterising the capability of a pollutant to propagate through the monitored rock environment, whereas the transportation medium is a liquid and/or gas. The main hydrogeological parameters of a rock environment include hydraulic conductivity, effective porosity, and hydraulic dispersivity.

Temelín NPP construction and described as required by the criterion laid down in Section 5(d) of Decree No. 215/1997 Coll. [L. 1] (see Section 2.6.7.11.2).

Hydraulic Conductivity[91]

The values of the groundwater flow velocity that were determined based on filtration parameters and the groundwater level slope on the construction site of Temelín NPP and its close surroundings are provided in [L. 113]. Tab. 130 presents the hydraulic gradient, average filtration coefficient and filtration velocity values for various areas in the surroundings of Temelín NPP. The values apply to the upper horizon of groundwater, specifically up to the depth of approximately 30 metres below the terrain.

The filtration and actual velocity values were determined also with the aid of tracing tests (see basic material [L. 57]) in the rock environment between boreholes RK-4 and RK-5 in the vicinity of the AASB. According to this basic material, the filtration coefficient is of the order of 10E-7, while the actual velocity is of the order of 10E-5.

Tab. 130 Groundwater flow velocities (taken from [L. 113]).

Area	Hydraulic gradient	Average filtration coefficient x 10^{-7} m.s^{-1}	Filtration velocity x 10^{-7} m.s^{-1}
NON-REDUCED VALUES			
Site area	0.009	3.8	0.0342
North-east	0.036	0.016	0.00058
South-east	0.016	0.021	0.00034
South-west	0.018	1.9	0.0342
REDUCED VALUES			
Site area	0.009	40	0.36
North-east	0.036	0.55	0.02
South-east	0.016	1.3	0.021
South-west	0.018	21.5	0.39

Effective Porosity

Basic materials, e.g. [L. 113], specify porosity values for disrupted samples of gneiss, sandy silt and silty sand, i.e. materials that form the most frequent filling of fracture zones and are present in eluviums and the Quaternary cover. Effective porosity[92] was determined by way of experiments (by tracing tests - see basic material [L. 57]) and expert estimates (see basic material [L. 113]). The basic materials indicate this parameter in units of %.

Hydraulic Dispersivity[93]

This parameter (longitudinal hydrodynamic dispersion coefficient α_L [m]) is also available in [L. 57]. The α_L coefficient was of the order of metres.

91 Hydraulic conductivity is defined as the ability of a porous environment to transmit liquid of a given kinematic viscosity when subjected to the effect of a hydraulic gradient (head). The measure of the hydraulic conductivity of water is the filtration coefficient.

92 Effective porosity is defined as the share of pore space, which is effective from a specific perspective, in the total rock volume, or as the share of pore space in the total rock volume, where fluids actually flow.

93 Dispersivity is a parameter of the porous environment, which defines the rate of spatial propagation of a dissolved substance.

Results of the Models of the Potential Propagation of Radionuclides by Groundwater

The characteristics of the rock environment with respect to the potential propagation and retardation of radionuclides are detailed in [L. 113]. The document specifies values of the retardation factor R_f and the distribution coefficient K_d for radionuclides ^{90}Sr, ^{137}Cs, and ^{60}Co and the environment of gneiss, weathered gneiss, sandy loam, loamy sand, and topsoil.

[L. 57] suggests that a number of factors on the Temelín NPP construction site impair vertical radionuclide migration. These factors particularly embrace the generally low permeability of the cover formations (low filtration coefficient values), filling-up of coarse-grained soils of the upper horizon on the site, and the stratified arangement of the soils with alternating permeable and less permeable beds. The possible effect of retarding and adsorbing mechanisms due to the share of clayey minerals, carbonates and iron oxides in the cover formation soils should also be considered favourable. Nevertheless, preferential pathways may be formed in areas with fracture permeability or in layers with higher intrinsic permeability (including artificial).

Even if investigations and tests aimed at ascertaining migration characteristics were not performed in areas S1 and S2, with a view to considerable similarity of the upper layer on the construction site of NPP1,2 and ETE3,4, analogical evaluation results may be expected. Therefore, it is possible to accept the conclusions provided in [L. 51] as the null, implicit hypothesis for the evaluation of the ETE3,4 construction site.

2.6.7.11.4 Criterion According to Section 5(f) (Decree No. 215/1997 Coll.)

The criterion relates to those hydrogeological foundation soil properties that may have an unfavourable effect on the foundation of the structures of a nuclear installation (ideal foundation conditions are above the groundwater level - see basic material [L. 186]). The placement of the structures, including their foundation structures, above the groundwater level also reduces the risk of direct contamination of groundwater by leaks of various harmful substances released during the construction and the operation of the nuclear installation.

Rock and soil "permeability" (intrinsic/fracture permeability) mentioned in the criterion is characterised by the filtration coefficient *k* expressed in $m.s^{-1}$. Based on the rock permeability classification [L. 114]), well permeable are rocks of class I (very high permeability), class II (high permeability) or class III (moderately high permeability) with filtration coefficient values ranging from 1.10^{-2} to 1.10^{-4} $m.s^{-1}$ (corresponds to the permeability coefficient from 10^{-9} to 10^{-11} m^2). In practice, such rocks are unconsolidated gravels, gravel sands and sands, often of glacial or fluvial origin.

Occurrence of "Well Permeable Rocks" on the ETE3,4 Construction Site

Soil properties, including hydraulic, were determined during investigations carried out on the main construction site of Temelín NPP (see e.g. basic material [L. 144], [L. 145]). These basic materials rate the underlying soils and eluviums as exhibiting "low permeability". In [L. 144], the average filtration coefficient is $k = 2.8.10^{-7}$ $m.s^{-1}$. Slightly higher values are anticipated in bodies of granitic rocks and beds of their eluviums. Such bodies and beds, however, are spatially limited and do not form a continuous horizon on the construction site.

According to the results of investigations in boreholes of the RK (radiation control) series, rock and soil permeability (i.e. filtration coefficient) in the zone of prevalent rock weathering ranges between 10E-6 and 10E-7 m.s^{-1} (see basic material [L. 51]). The same basic material evaluates permeability in the zone of slightly weathered and fresh rocks at depths of dozens of metres below the terrain. The filtration coefficient in this environment of fracture permeability is of the order of 10E-7 to 10E-8.

At present, as indicated by the results of the supplementary engineering geological survey [L. 165]), the most intensely saturated beds are largely located at old excavation sites, near covered underground structures, etc., where mainly the backfill material of sandy or sandy-silty character is saturated.

A similar effect may be observed in the vicinity of the buildings of the part of Temelín NPP that is currently in operation, where the level of groundwater in these saturated backfills must be artificially lowered.

Evaluation of Groundwater Levels on the ETE3,4 Construction Site

Information on groundwater levels was obtained from overviews of average monthly levels in the monitoring boreholes at Temelín NPP (data of the Environmental Division of Temelín NPP). The levels are automatically recorded by the NOEL 2000 system.

Groundwater levels in the monitoring boreholes in the site area of Temelín NPP (RK-2, HV-615, RK-25) range from 5 to 9 m below the terrain. It should be noted, however, that the levels are considerably influenced by withdrawals through pumping wells. It is possible to observe steeply dropping groundwater levels immediately after precipitation, which is probably due to the activation of the pumps of the drainage system. Groundwater withdrawals are also the probable cause of the fluctuating water level in borehole HV-4C located outside the site area of Temelín NPP, where it varies between 1.5 and 5.5 metres below the terrain.

The groundwater level in other boreholes responds to precipitation.[94] The evaluation presented in [L. 179] implies that outside the site area of Temelín NPP, the range of groundwater levels in the monitoring boreholes is rather wide, i.e. from 0 to 7 metres below the terrain. The most frequent level is between 2 and 2.5 metres below the terrain.

In the site area of Temelín NPP, groundwater levels are around 6.5 metres below the terrain as a result of pumping. The assessment of this parameter in areas S1 and S2 is very limited by the fact that no monitoring boreholes are located there. Available data on groundwater depths are from July 2010[95] (see basic material [L. 165]). The groundwater level on area S1 was then 4 metres below the terrain and 4.5 metres below the terrain on the adjacent area S2.

The water level on area S2 is probably influenced by two factors. The first is the existence of the extensive storm sewer system of the structures belonging to the former constructional plant. The second factor is the existence of an artificial water basin (wetland) in the eastern tip of area S2. Even if the water basin does not actually

[94] See data of the Czech Hydrometeorological Institute - Department of Nuclear Power Plant Observatories, observatory in Temelín. Also used in the study of Smítek J. (2011): The impact of an increase in the heat output of Temelín NPP units 1 and 2 by 4% or 5.8% on the microclimate in the surroundings of Temelín Nuclear Power Plant - average monthly precipitation from 2004 to 2010.

[95] July 2010 was above average in terms of precipitation volumes, the data measured in this period may be therefore considered suitable for the purpose of estimating the maximum groundwater level.

fulfil any drainage function, it draws groundwater from its surroundings and it may thus influence the groundwater levels on both area S2 and the adjoining area S1.

2.6.7.11.5 Summarized Evaluation

The results of the evaluation embracing the criteria and requirements specified in Section 7.11 allow the following conclusions to be drawn:

[1] Although the precipitation-runoff conditions are rather complicated on the construction site as a result of numerous interventions in the natural flow of groundwater, a reliable model of the runoff and the direction of the flow of groundwater in the upper horizon may be created. Therefore, it is possible to monitor and predict groundwater behaviour on the site designated for the siting of ETE3,4. This means that the construction site is not in conflict with the criterion stipulated in Section 5(d) of Decree No. 215/1997 Coll. [L. 1].

[2] No significant soil beds that may be considered as "well" permeable are present on the construction site. The level of groundwater on the ETE3,4 construction site and its close surroundings lies at a maximum of 4 to 4.5 metres below the terrain. The terrain represents the level of rough ground shaping.

[3] Drainage of the ETE3,4 construction site and the conceivable lowering of the groundwater level at the site of construction below the limit specified by the criterion in Section 5(f) is technically feasible. The specific design of an effective drainage system will be a part of the subsequent project execution phases. Moreover, all substructures will have a sufficient insulation system. The requirement concerning the proposal of measures preventing the inundation of excavation pits was also included in the conditions for the contractor. This means that the construction site is not in conflict with the criterion stipulated in Section 5(f) of Decree No. 215/1997 Coll. [L. 1].

[4] It is possible to confirm that investigations aimed at gaining an understanding of the migration and retention characteristics of soils, the dispersion characteristics of aquifers, and the physical and physico-chemical properties of underground rocks and soils were carried out. Likewise models of the potential propagation of radionuclides by groundwater were developed. Even though these investigations and analyses were performed during the evaluation of the operational safety of NPP1,2, they may be considered valid also with respect to the initial evaluation of the ETE3,4 construction site. This means that the requirements stipulated by Paragraphs 4.8 and 4.9 of the IAEA Safety Requirements No. NS-R-3 [L. 6] are complied with.

2.6.7.12 CRITERIA AND REQUIREMENT ACCORDING TO SECTION 7.12

2.6.7.12.1 Specification of Hazards Grouped in Section 7.12

Section 7.12 (see Tab. 117) contains the criterion defined by Section 5(e) of Decree No. 215/1997 Coll. [L. 1], which involves the evaluation of groundwater aggressiveness on the nuclear installation construction site. The evaluation performed in compliance with this section also incorporates the requirement defined in Paragraph 3.43 of the IAEA Safety Requirements No. NS-R-3 [L. 6].

2.6.7.12.2 Requirement According to Paragraph 3.43 (NS-R-3) - Part 2

The relevant paragraph lays down a general requirement consisting in an evaluation of the groundwater regime and the chemical properties of groundwater.

The chemical composition of groundwater on the construction site of Temelín NPP is observed within operational monitoring (see basic material [L. 89]). The quality of groundwater in the site area of Temelín NPP is monitored by boreholes of the RK series, which are located close to important structures and facilities, e.g. near the Nuclear auxiliary service building, near the Essential water cooling spray basin, near the Chemicals storage, below the Dieselgenerator station of unit 1, behind the Turbine hall of unit 1, and near the Cooling towers (see also [L. 113]). Additional 9 boreholes serve for the monitoring of the wider surroundings of Temelín NPP. These boreholes intercept deep, middle and shallow aquifer in the vicinity of the power plant.

In 2012, control samples of ground and surface waters were collected in the surroundings of the ETE3,4 construction site (see basic material [L. 179]). The location of the sample collection sites is shown on the map in Fig. 58. The analysis results are summarised in Tab. 131.

Tab. 131 Results of the chemical analyses of groundwater in the surroundings of the ETE3,4 construction site.

Borehole/Sampling point		RK-2	RK-26	HV 1002	HV 1005	VZ-1	VZ-4	VZ-5	VZ-6
Date of sampling		03/04/12	03/04/12	03/04/12	03/04/12	04/04/12	04/04/12	04/04/12	04/04/12
Indicator	**Unit**								
Water reaction (pH)	pH	6.8	6.9	7.4	6.9	5.9	8.1	7.6	7.9
BNC 8.3	mmol/l	0.71	0.75	0.28	0.50	0.75	1.10	0.43	0.59
ANC 4.5	mmol/l	2.30	3.10	1.80	2.60	0.61	4.60	2.00	2.90
BNC 4.5	mmol/l	0	0	0	0	0	0	0	0
ANC 8.3	mmol/l	0	0	0	0	0	0	0	0
Conductivity	mS/m	57	48	25	24	30	56	35	46
Hardness (Ca+Mg)	mmol/l	1.7	2.3	1.0	1.1	1.3	2.8	1.6	1.9
COD-Mn	mg/l O2	<0.40	<0.40	0.64	0.51	1.4	1.5	1.8	1.5
Total Dissolved solid	mg/l	400	430	220	280	220	530	290	390
Ammonia ions	mg/l	<0.050	<0.050	0.070	<0.050	<0.050	<0.050	0.060	<0.050
Lithium - Li	mg/l	0.012	0.038	0.040	0.027	0.0040	0.0080	0.004	0.013
Sodium - Na	mg/l	36	18	13	12	9.5	15	12	22
Potassium - K	mg/l	9.3	4.4	4.6	3.7	4.8	17	5.2	8.3
Magnesium - Mg	mg/l	14	22	12	13	9.3	17	11	15
Calcium - Ca	mg/l	45	57	20	22	35	83	46	52
Total iron	mg/l	0.75	1.6	4.3	41	0.48	<0.10	0.93	0.14
Manganese - Mn	mg/l	<0.050	0.25	0.30	0.55	<0.050	<0.050	0.34	<0.050
Fluorides	mg/l	0.39	0.52	0.56	0.45	0.12	1.1	0.48	0.74
Chlorides	mg/l	91	8.2	7.2	4.7	27	10	13	29
Nitrites	mg/l	<0.010	<0.010	<0.010	<0.010	<0.010	<0.010	0.010	0.010
Nitrates	mg/l	9.5	<1.0	<1.0	1.6	40	8.3	13	12
CO2	mg/l	24	15	4.6	22	28	-	9.5	-

Borehole/Sampling point Date of sampling		RK-2 03/04/12	RK-26 03/04/12	HV 1002 03/04/12	HV 1005 03/04/12	VZ-1 04/04/12	VZ-4 04/04/12	VZ-5 04/04/12	VZ-6 04/04/12
Indicator	**Unit**								
aggressive									
Hydrogen carbonates	mg/l	140	190	110	160	37	280	120	180
Carbonates	mg/l	0	0	0	0	0	0	0	0
Sulphates	mg/l	45	120	44	7.3	49	84	63	63
Phosphates	mg/l	0.27	0.12	<0.10	0.13	0.15	0.14	0.10	1.0
Silicon - Si	mg/l	8.5	11	4.7	10	5.3	8.0	5.4	6.6

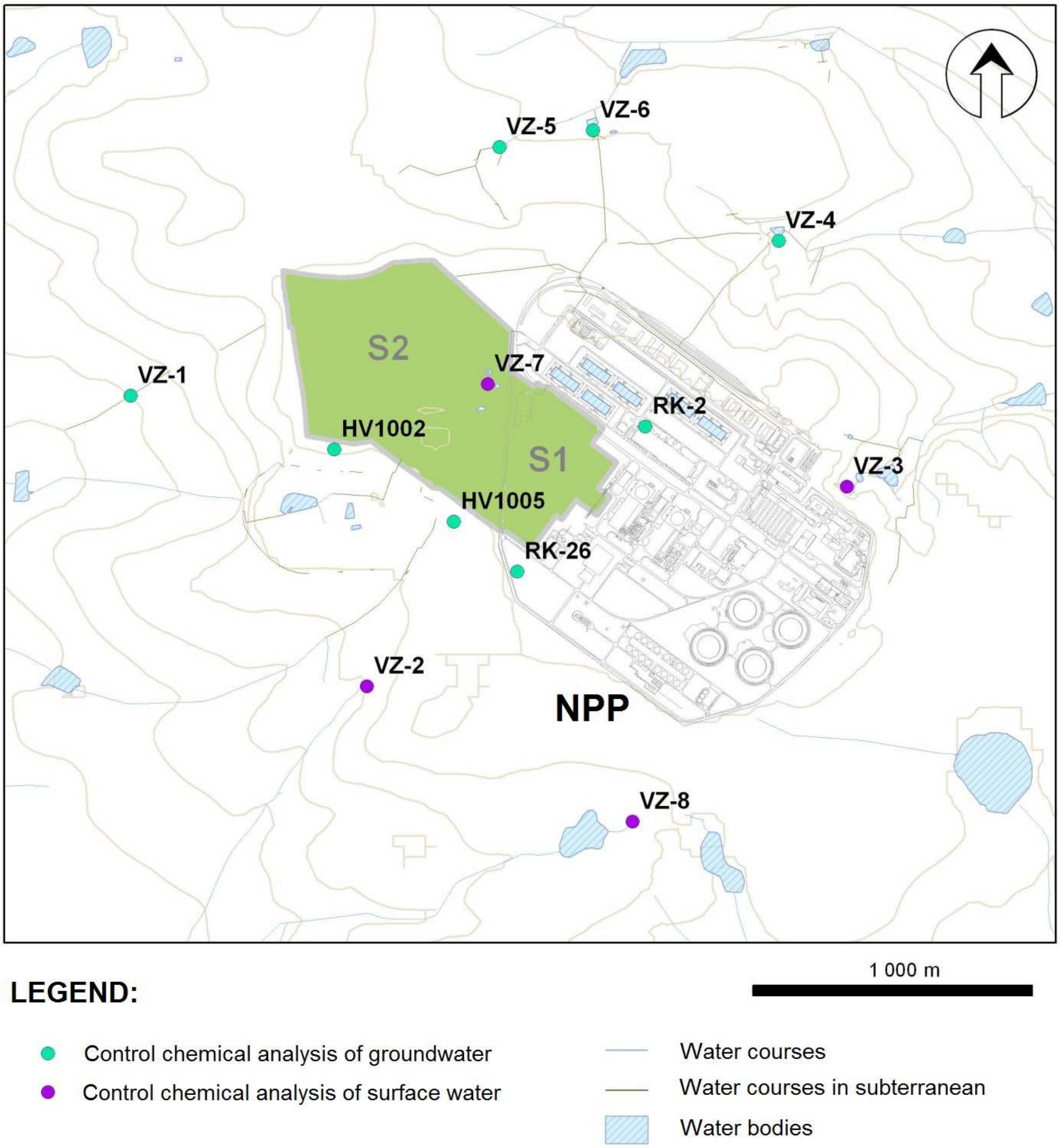

Fig. 58 Sampling points of control samples of ground and surface waters in the surroundings of the ETE3,4 construction site

Long-term variations in the chemism of groundwater may be evaluated with the use of the basic material provided in the annual reports on the "Monitoring and Evaluation of the Quality of Surface and Groundwaters and Their Change in Connection with the Impact of the Construction and Operation of the Temelín Nuclear Power Plant on Its Surroundings" (see e.g. 2010 annual report - [L. 89]).

Based on the analysis results, the groundwaters on the ETE3,4 construction site are moderately mineralised, slightly acidic or neutral, often of the HCO_3-SO_4-Ca-Mg type. Increased contents of Na, Cl, and NO_3 ions were detected in some samples and these indicate possible anthropogenic pollution.

The chemism of surface waters is also monitored. The results are also presented in the annual reports on the "Monitoring and Evaluation of the Quality of Surface and Groundwaters and Their Change in Connection with the Impact of the Construction and Operation of the Temelín Nuclear Power Plant on Its Surroundings". The results of control measurements are specified in [L. 179].

2.6.7.12.3 Criterion According to Section 5(e) (Decree No. 215/1997 Coll.)

The criterion addresses the potential reduction of the safety of structures due to the effects of aggressive groundwater that is likely to occur with every additional protective structural element (e.g. insulation), the failure of which may be reasonably anticipated (see basic material [L. 186]).

The essence of water aggressiveness lies in the ability of groundwater to disrupt and deteriorate materials, most often building materials (e.g. concrete, metals), with which it comes into contact. One of the consequences of groundwater aggressiveness is the corrosion of materials (see [L. 153]).

In general, the corrosion of mortar and concrete caused by the effects of aggressive impact water may be divided into several types:

- Corrosive effects of water due to low mineralisation (leaching aggressiveness);
- Corrosive effects of water due to its low pH (acidic water);
- Corrosive effects of water due to the content of sulphate ions (sulphate aggressiveness);
- Corrosive effects of water due to the content of aggressive carbon dioxide;
- Corrosive effects of water due to the high content of magnesium ions;
- Corrosive effects of water due to the high concentration of ammonia nitrogen;
- Corrosive effects of water due to the content of other agents (e.g sulphane and its ion forms).

The basic evaluation of ground and surface water aggressiveness is regulated by CSN EN 206–1 "Concrete – Part 1: Specification, Properties, Production and Conformity". The evaluation also makes use of findings from the field of hydrochemistry[96,97,98] (see e.g. lit. [L. 157]). Pursuant to the above specified standard,

[96] The effects of slightly mineralised water, which are not regulated by the standard, are rated according to the concentration of hydrogencarbonates in the water (ANC 4.5). Slightly aggressive water is water with ANC 4.5 < 2.0 mmol.l-1, i.e. its total mineralisation is approx. 200 mg.l-1. The corrosive effects of sulphane are triggered if its concentration in the water is equal to 1 mg.l-1 see lit. [L. 157].

the classification of the aggressiveness of the water environment recognizes three levels that are based on the values of the standard indicators presented in Tab. 132.

Tab. 132 Degree of aggressiveness of groundwater according to CSN EN 206–1

Chemical characteristics		Reference testing method	Classes of chemical attack		
			Slightly aggressive	Moderately aggressive	Highly aggressive
			XA1	XA2	XA3
SO_4^{2-}	[mg/l]	EN 196-2	≥ 200 and ≤ 600	> 600 and ≤ 3000	> 3000 and ≤ 6000
pH	[-]	ISO 4316	≥ 5.5 and ≤ 6.5	≥ 4.5 and < 5.5	≥ 4.0 and < 4.5
Aggressive CO_2	[mg/l]	prEN 13577:1999	≥ 15 and ≤ 40	> 40 and ≤ 100	> 100 up to saturation
NH_4^+	[mg/l]	ISO 7150-1 / -2	≥ 15 and ≤ 30	> 30 and ≤ 60	> 60 and ≤ 100
Mg^{2+}	[mg/l]	ISO 7980	≥ 300 and ≤ 1000	> 1000 and ≤ 3000	> 3000 up to saturation

The assessment of groundwater aggressiveness was a part of the constructional geological surveys that were conducted in all phases of the Temelín NPP construction site verification process.

The results of the groundwater analyses imply that, within the wording of CSN EN 206–1, the environment on the construction site of Temelín NPP is slightly aggressive (exposure class XA1) with a view to the content of aggressive CO_2, locally and moderately aggressive (exposure class XA2) – see e.g. basic material [L. 156]. With respect to the content of sulphate ions, the environment is slightly aggressive (exposure class XA1) - cf. basic material [L. 165]. Concurrently, the groundwater environment may locally fulfil the preconditions for the development of aggressiveness arising from water acidity.

The results of the analyses of groundwater from the Spent fuel storage (SFS) construction site (see Tab. 133) and from areas S1 and S2 (see Tab. 135) are provided in the tables below. The sample collection sites are shown on the map in Fig. 59.

Tab. 133 Aggressiveness of groundwater (SFS construction site) according to the results of investigations conducted in 2006 [L. 155]).

Borehole Date of sampling		SVP-4 27/07/2006	SVP-5 27/07/2006	RK-26 09/08/2006
Indicator	Units			
Water reaction (pH)	pH	6.5	6.2	7.3
BNC 8.3 (acidity)	mmol/l	0.7	1.9	1.1
ANC 4.5 (alkalinity) m	mmol/l	0.9	0.8	4.1
BNC 4.5 (apparent acidity)	mmol/l	0	0	0
ANC 8.3 (apparent alkalinity) p	mmol/l	0	0	0

[97] In terms of acidic effects, groundwater is not aggressive if the water reaction (pH) is not less than 7.0 and temporary hardness < 4.3 mmol.l-1 or 6.7 and temporary hardness > 4.3 mmol.l-1 see lit. [L. 157].

[98] With pH > 8.3, the groundwater cannot contain free (or aggressive) CO_2.

Conductivity	mS/m	53.6	22.9	58.6
Hardness (Ca+Mg)	mmol/l	2.3	0.7	2.8
CO2 aggressive	mg/l	41	58	5.5
Ammonia ions	mg/l	<0.05	<0.05	0.11
Magnesium - Mg	mg/l	21	2	18
Calcium - Ca	mg/l	58	26	82
Sulphates	mg/l	200	65	230
Chlorides	mg/l	14	6.1	5.5

Aggressiveness XA1 XA2 XA3

The resultant aggressiveness rating is emphasized in the table by a background, the colour of which corresponds to the scale shown below the table. In addition, some samples meet the preconditions for aggressiveness arising from low water mineralisation (cf. footnote 96) and low pH (cf. footnote 97). The distinct differences in the chemical composition of the groundwater indicate that the water-bearing horizons are non-coherent. They are primarily formed along landfill bases or in layers of sandy eluviums and disrupted (jointed) rocks.

Additional data on the concentration of aggressive CO_2 are available through a sample of water collected at the drainage outlet (VZ-5) north of area S2 (the outlet mouths into Paleček Creek [Palečkův potok]) and at the outlet of damaged melioration west of the waste disposal facility in Temelínec (VZ-1). The concentration of aggressive CO_2 detected in sample VZ-1 was 28 mg.l^{-1} (exposure class XA1), while only 9.5 mg.l^{-1} in sample VZ-5. Moreover, the pH value determined for sample VZ-1 corresponds to a slightly aggressive environment belonging to exposure class XA1.

Groundwater aggressiveness was also investigated within the study [L. 179]. Samples were collected from 4 boreholes located in the vicinity of the ETE3,4 construction site Fig. 59. The analysis results are detailed in Tab. 136.

Tab. 134 Aggressiveness of groundwater on area S1 according to the results of investigations carried out in 2010 [L. 165].

Borehole		SE001	SE002	SE003
Date of sampling		**03/08/2010**	**21/07/2010**	**03/08/2010**
Indicator	**Units**			
Water reaction (pH)	pH	6.8	6.7	6.7
BNC 8.3 (acidity)	mmol/l	0.5	1.6	0.25
ANC 4.5 (alkalinity) m	mmol/l	2.7	3.8	1.7
BNC 4.5 (apparent acidity)	mmol/l	0	0	0
ANC 8.3 (apparent alkalinity) p	mmol/l	0	0	0
Conductivity	mS/m	26	100	24
Hardness (Ca+Mg)	mmol/l	1.6	4.4	1.0
CO2 aggressive	mg/l	3.1	29.0	14.0
Ammonia ions	mg/l	0.24	<0.05	0.08
Magnesium - Mg	mg/l	20	34	9
Calcium - Ca	mg/l	29	120	25
Sulphates	mg/l	48	430	34
Chlorides	mg/l	17	11	15

Aggressiveness XA1 XA2 XA3

Tab. 135 Aggressiveness of groundwater in area S2 according to the results of investigations carried out in 2010 [L. 165].

Borehole		CT101	CT102	CT103	CT106	SE004
Date of sampling		01/07/2010	28/07/2010	22/07/2010	12/07/2010	09/07/2010
Indicator	Units					
Water reaction (pH)	pH	8.2	7.7	7.5	6.7	7.1
BNC 8.3 (acidity)	mmol/l	0.3	0.35	0.1	0.2	1.1
ANC 4.5 (alkalinity) m	mmol/l	8.4	0.97	2.3	0.58	4.4
BNC 4.5 (apparent acidity)	mmol/l	0	0	0	0	0
ANC 8.3 (apparent alkalinity) p	mmol/l	0	0	0	0	0
Conductivity	mS/m	97	64	47	33	74
Hardness (Ca+Mg)	mmol/l	4.2	2.7	1.3	0.93	3.5
CO2 aggressive	mg/l	<1.0	6.6	<1.0	36	Non-stab.
ammonia ions	mg/l	<0.05	0.19	0.15	0.12	<0.05
Magnesium - Mg	mg/l	41	33	12	6.7	41
Calcium - Ca	mg/l	65	48	32	26	72
Sulphates	mg/l	160	270	120	27	150
Chlorides	mg/l	10	15	13	41	20

Aggressiveness XA1 XA2 XA3

Tab. 136 Groundwater aggressiveness assessment based on the results of investigations carried out in 2012 [L. 179]).

Borehole		RK-2	RK-26	HV-1002	HV-1005
Date of sampling		03/04/2012	03/04/2012	03/04/2012	03/04/2012
Indicator	Units				
Water reaction (pH)	pH	6.8	6.9	7.4	6.9
BNC 8.3 (acidity)	mmol/l	0.71	0.75	0.28	0.5
ANC 4.5 (alkalinity) m	mmol/l	2.3	3.1	1.8	2.6
BNC 4.5 (apparent acidity)	mmol/l	0	0	0	0
ANC 8.3 (apparent alkalinity) p	mmol/l	0	0	0	0
Conductivity	mS/m	57	48	25	24
Hardness (Ca+Mg)	mmol/l	1.7	2.3	1.0	1.1
CO2 aggressive	mg/l	24	15	4.6	22
Ammonia ions	mg/l	<0.050	<0.050	0.070	<0.050
Magnesium - Mg	mg/l	14	22	12	13
Calcium - Ca	mg/l	45	57	20	22
Sulphates	mg/l	45	120	44	7.3
Chlorides	mg/l	91	8.2	7.2	4.7

Aggressiveness XA1 XA2 XA3

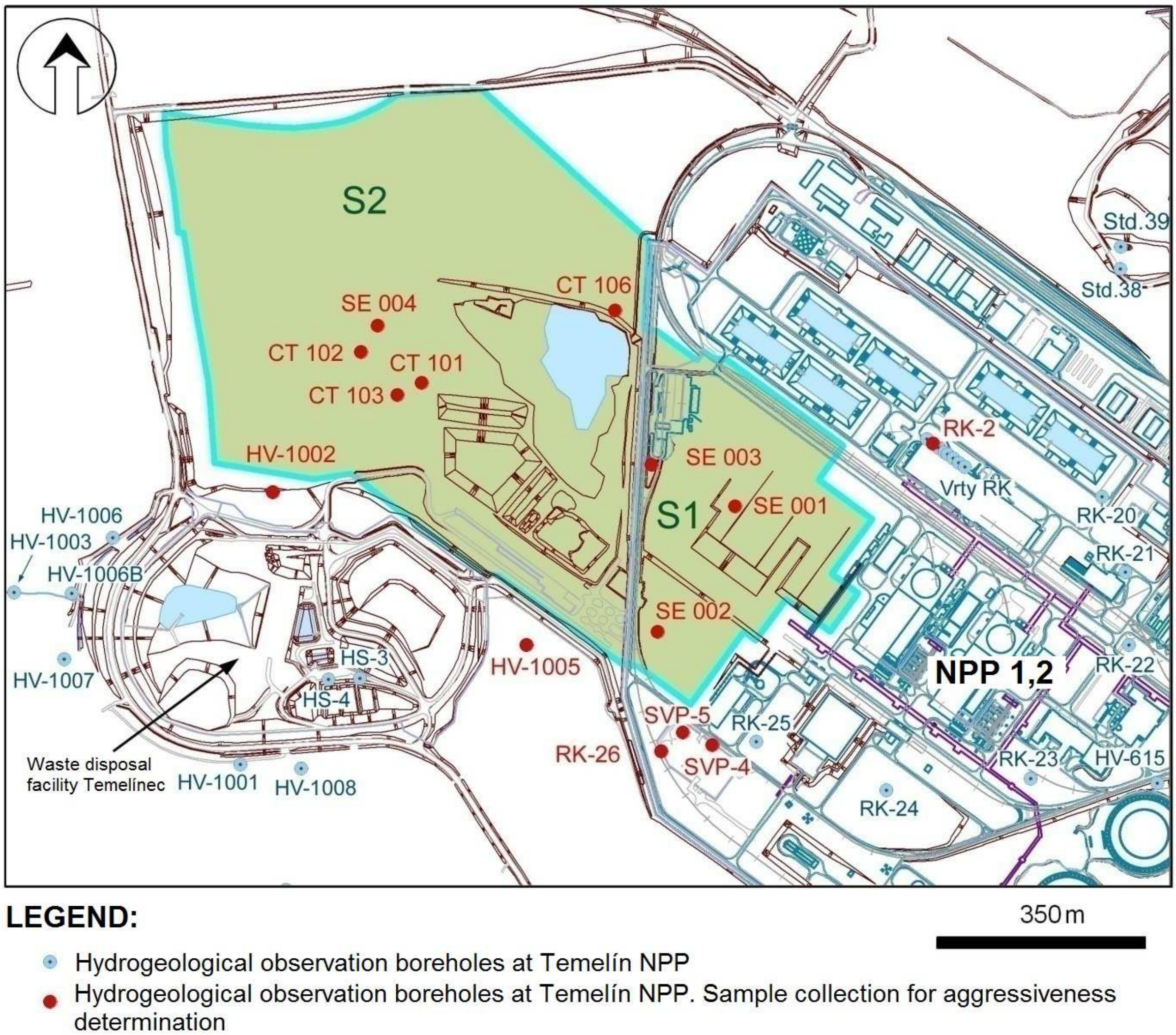

Fig. 59 Map of the collection sites of samples for the purpose of groundwater aggressiveness assessment.

2.6.7.12.4 Summarized Evaluation

With a view to the above specified findings, the site vicinity of ETE3,4 is not in conflict with the exclusion criterion defined in Section 5(e) of Decree No. 215/1997 Coll. [L. 1]. Therefore, it is possible to conclude that:

[1] The chemical composition of groundwater (and surface water) is monitored on a regular basis at specific points of the monitoring system, which covers the entire construction site of Temelín NPP and its close surroundings. The analysis results are evaluated and published in annual reports on the "Monitoring and Evaluation of the Quality of Surface and Groundwaters and Their Change in Connection with the Impact of the Construction and Operation of the Temelín Nuclear Power Plant on Its Surroundings".

[2] The level of aggressiveness of the groundwater on the construction site of Temelín NPP is commonly encountered in a crystalline rock environment. It primarily manifests itself in the exposure class XA1 (pursuant to CSN EN 206–1) in terms of the content of aggressive CO_2 and sulphate ions (rarely in the exposure class XA2 in case of aggressive CO_2). Weak aggressiveness is also caused by the acidity of the water.

[3] The protection of the foundation structures against the aggressive effects of groundwater is technically feasible and the presentation of a proposal for such a solution will be included among the requirements placed on the contractor.

2.6.8 CONCLUDING EVALUATION

2.6.8.1 FULFILMENT OF THE CRITERIA AND REQUIREMENTS

The following conclusions may be made based on the analyses and evaluations of the relevant criteria and requirements (see Section 2.6.7):

- With respect to the hazards arising from the occurrence of earthquakes, no conflicts with the exclusion criterion stipulated in Section 4(e) or with the conditional criterion defined in Section 5(c) of Decree No. 215/1997 Coll. [L. 1] were identified. Likewise, all relevant requirements specified in the applicable paragraphs of the IAEA Safety Requirements No. NS-R-3 [L. 6]
- In relation to the hazards arising from the movement and seismic activity of faults, no conflicts with the exclusion criteria stipulated in Sections 4(f) and 4(i) of Decree No. 215/1997 Coll. [L. 1] were identified. The assessment of these hazard jeopardy was performed in compliance with the relevant requirements specified by the applicable paragraphs of the IAEA Safety Requirements No. NS-R-3 [L. 6].
- In terms of the engineering geological hazards, the site vicinity or the construction site are not in conflict with any of the relevant exclusion criteria stipulated in Section 4 of Decree No. 215/1997 Coll. [L. 1] (namely Section 4(c), Section 4(d), Section 4(g), Section 4(h), Section 4(k), Section 4(n), Section 4(o)). The site vicinity is not encumbered with any requirements in terms of overcoming unfavourable conditions on the territory that arise from any of the relevant conditional criteria (namely Section 5(a), Section 5(b) of thereof. With respect to the engineering geological hazards, all relevant requirements set out in the IAEA Safety Requirements No. NS-R-3 [L. 6] have been complied with.
- In relation to hydrogeological hazards, the site vicinity is not in conflict with the exclusion criterion stipulated in Section 4(j). In this respect, however, attention should be given to the drainage of the ETE3,4 construction site. The conceivable occurrence of groundwater in excavation pits or usual levels of groundwater aggressiveness (or phenomena associated with the conditional criteria defined in Section 5(e) and Section 5(f) of Decree No. 215/1997 Coll. [L. 1]) may be resolved on the ETE3,4 construction by means of technical measures. Their proposal, which should be based on the supplementary constructional hydrogeological survey, will be incorporated in the subsequent phases of the ETE3,4 project execution, depending on the layout of the structures and facilities and the depth and method of their foundation.
- With respect to hydrogeological hazards, all relevant requirements laid down by the IAEA Safety Requirements No. NS-R-3 [L. 6] have been complied with.

2.6.8.2 STARTING POINTS FOR SUBSEQUENT PROJECT EXECUTION PHASES

The ascertained engineering geological conditions on the construction site suggest the following starting points for the project:

- With a view to the determined seismic load for the near region of ETE3,4 (see Section 2.6.7.1.6), a seismic design with the minimum SL-2 value of 0.1 g may be accepted in compliance with Paragraph 2.11 of the IAEA Specific Safety Guide No. SSG-9 [L. 14].
- The main construction site of Temelín NPP (see footnote 37) is situated on a uniform and stable geological block that is disrupted by tectonic discontinuities only to a very small extent.
- The foundation conditions in relation to the presumed localization of the structures and facilities of ETE3,4 are currently characterised within the extent of a preliminary survey with the use of the previous in-depth engineering geological survey of the main construction site.
- The available findings on the geological conditions on the ETE3,4 construction site are supported by the results of exploratory drilling, in particular by the data obtained from deeper core holes drilled to the depth of 50 metres. These findings imply that prevalent weathering of the underlying rock mass up to the elevation of 495 metres above sea level should be anticipated in the area of the ETE3,4 Units. The same applies (to a somewhat greater extent) also for the area of the contemplated placement of the cooling towers. In general, however, the elevation of 495 metres above sea level and the modulus of deformation of the rocks over 100 MPa should be taken into account. At this level, the shear (S) wave velocity greater than 1100 m.s-1 (for details see Section 2.6.2.3) may be envisaged.
- The foundation soil on the ETE3,4 construction site consists of solid rock (for details see Section 2.6.2.3); this means that the foundation conditions may be considered as simple (within the meaning of Article 20(a) of CSN 73 1001).[99] The crucial structures of ETE3,4 will be adjudged as demanding structures.
- All structures and facilities of ETE3,4 may be founded with the use of standard methods.
- The ascertained shear wave velocity value (see Tab. 113) indicates that the ETE3,4 construction site is a type 1 site within the meaning of Paragraph 3.1 of the IAEA Safety Guide No. NS-G-3.6 [L. 13] and it is therefore not necessary to perform a dynamic analysis of the seismic response for the structures.
- Measures should be implemented on the ETE3,4 construction site aimed at draining the construction site and the project should also include hydroinsulation of the foundation structures (see Section 2.6.7.11.5).
- The parameters of the groundwater on the ETE3,4 construction site should be monitored. The NNP3, 4 monitoring system should expand the current system for NPP1,2.

[99] The survey and its evaluations were performed in compliance with the then valid standard. Similarly to the original standard, the new European standard CSN EN 1997 introduces geotechnical categories for the classification of structures according to the complexity of the foundation conditions and the constructed structure. The foundation design is subsequently selected according to this classification. The classification will be reviewed to comply with the newly standard for the purposes of contractor design project.

2.7 RADIATION SITUATION AT THE SITE

2.7.1 SCOPE OF THIS SECTION

This chapter contains A description of the current radiation situation at the Temelín site and assessment of the feasibility of siting and of the acceptability of operating a new nuclear source from the aspect of radiation effects on the population and environment.

2.7.2 SUMMARY OF FACTS

2.7.2.1 EXPOSURE OF INDIVIDUALS FROM THE CRITICAL GROUP OF THE PUBLIC

The contribution of current operation of the Temelín nuclear power plant to the radiation burden of the population in the surroundings is so low that it cannot be determined by direct measurement. Virtually the only feasible way to assess this burden is by determining the effective doses and committed doses for individuals from the critical population group using the authorized RDETE computer code based on a balance of the gaseous and liquid radionuclide discharges.

Tab. 137 and Tab. 138 document current contributions from the operation of the Temelín ETE1,2 reactor units for the various age groups in towns and villages which lie in the prevailing downwind direction and are most exposed to the effect of gaseous discharges (Litoradlice, Zvěrkovice) and in towns and villages which are exposed to the effect of liquid discharges (Pašovice and Neznašov).

Tab. 137 Effective doses and dose commitments received by representative individuals in different age groups of the population from atmospheric discharges, 2005 – 2011

Age group [years]	[nSv]						
	2005	2006	2007	2008	2009	2010	2011
0 to 1	99	45	37	23	10	11	18
1 to 2	171	50	47	29	11	13	23
2 to 7	178	53	50	30	12	14	23
7 to 12	188	52	49	30	11	13	23
12 to 17	171	51	48	29	10	12	20
Adults	184	52	49	30	10	12	20

Remark: Radionuclide C-14 contributes a predominating fraction (about 95%) to the calculated effective dose commitment. The substantial difference between the current levels and the 2005 levels is due to a change in the calculation methodology.

Tab. 138 Effective doses and dose commitments received by representative individuals in different age groups of the population from discharges into watercourses, 2005 – 2011

Age group [years]	[nSv]						
	2005	2006	2007	2008	2009	2010	2011
0 to 1	228	396	301	584	684	556	821
1 to 2	175	301	229	452	540	440	638
2 to 7	162	278	211	420	615	501	729

Age group [years]	[nSv]						
	2005	**2006**	**2007**	**2008**	**2009**	**2010**	**2011**
7 to 12	124	210	160	325	480	392	558
12 to 17	101	168	128	265	389	318	448
Adults	147	243	184	372	543	444	662

Remark: Radionuclide H-3 contributes a predominating fraction (95%) to the calculated effective dose commitment. The differences in the levels between the years are due both to differences in the tritiated water discharges and differences in water flow through the Vltava.

The data in the table give evidence that:

- The effective dose and committed effective dose arising from radionuclides released into the environment through discharges into air are below the regulatory effective dose / committed effective dose limit for an individual in the critical population group, i.e. 40 μSv per annum, laid down by SÚJB Decision No. 28718/2007 of 29 November 2007.
- The effective dose and committed effective dose from radionuclides released into the environment through discharges into watercourses are below the regulatory effective dose and committed effective dose limits for an individual from the critical population group, i.e. 3 μSv per annum, laid down by SÚJB Decision SÚJB/OROPC/26161/2009 of 1 December 2009.

2.7.2.2 MONITORING OF DISCHARGES

The following sources of gaseous discharges are monitored at the Temelín nuclear power plant:

- Double-shaft stack at main power generation unit 1
- Double-shaft stack at main power generation unit 2
- One ventilation stack at the building of active auxiliary operations
- Secondary coolant circuit bleed at main power generation unit 1
- Secondary coolant circuit bleed at main power generation unit 2

Liquid discharge monitoring at the Temelín NPP is based on balance measurements of the various radionuclides in the control tanks prior to discharge into the wastewater collecting basin and subsequently into the sunken step at Kořensko. Those tanks which are automatically monitored by radiation control instrumentation must not be drained without the consent of the shift engineer. If exceeding of the preset reference activity levels of the liquid discharges is detected, the valves are shut automatically and the tank discharge process is discontinued.

Water activity in the collecting basin at the Temelín nuclear power plant wastewater channel is continuously monitored by means of a system which, in addition, collects a proportionally pipetted sample for off-line spectrometric or radiochemical analyses of the water discharged.

2.7.2.3 METHODS EMPLOYED TO MONITOR THE RADIATION SITUATION AT THE TEMELÍN SITE

The radiation situation monitoring in the surroundings of the Temelín NPP is governed by the "Programme of Monitoring of the Surroundings of the Temelín NPP" [**P. 48**]. Discharges are monitored to the extent specified in ČEZ's methodological document "Programme of Monitoring of Discharges from the Temelín nuclear power plant" [**P. 49**], approved by the SÚJB (Decision No. 17893/2007 of 20 June 2007).

All balance and verified measurements were performed by using specified meters, validated by the Czech Metrology Institute – Inspectorate of Ionising Radiations or by Accredited Calibration Laboratory No. 2245.2, ČEZ, a. s., Dukovany nuclear power plant. That accredited laboratory also calibrates photon dose equivalent rate measurements using thermoluminescent dosimeters.

Radionuclide determination in waters and soil is verified by intercalibration performed by the Accreditation Centre, ASLAB hydroanalytical laboratories. Determination of radionuclides in aerosols (gamma nuclides, $^{239,240}Pu$ and ^{90}Sr), in fallout (gamma nuclides), in waters (gamma nuclides, ^{3}H and ^{90}Sr) and determination of the photon dose equivalent H_x by using thermoluminescent dosimeters (TLD) are verified through analyses performed by the National Radiation Protection Institute (SÚRO) in Prague within the "Radiation Monitoring Network of the Czech Republic", whose operation is governed by Decree No. 319/2002 Coll. [**L. 25**].

The Radiation Monitoring Laboratory for the surroundings of the Temelín nuclear power plant was granted Certificate of Accreditation No. 408/2008 on 29 September 2008 (the Testing Laboratory number is 1241.4). The Certificate was issued by the Czech Accreditation Institute based on the laboratory's compliance with the accreditation criteria laid down by ČSN EN ISO/IEC 17025:2005 for the following methods:

- 1. Determination of volumetric activities of radionuclides by laboratory gamma spectrometry – aerosols in air
- 2. Determination of volumetric activities of radionuclides by laboratory gamma spectrometry – natural waters
- 3. Determination of volume activities of radionuclides by laboratory gamma spectrometry – milk
- 4. Determination of the (photon) dose equivalent rate in a gamma field by using an RSS-131 high-pressure ionisation chamber, at monitoring sites within the nuclear power plant area and in its surroundings
- 5. Determination of volume activity of tritium by laboratory liquid scintillation spectrometry of beta radiation – gaseous discharges
- 6. Determination of volume activity of C-14 by laboratory liquid scintillation spectrometry of beta radiation – gaseous discharges
- 7. Determination of the total volume alpha activity in waters – surface waters and wastewaters
- 8. Determination of the total volume beta activity in waters – surface waters and wastewaters

- 9. Determination of volume activity of tritium by laboratory liquid scintillation spectrometry of beta radiation – drinking water, ground water, surface waters, rainwater and snow water and process waters
- 10. Determination of volume activity of tritium by laboratory liquid scintillation spectrometry of beta radiation – wastewaters

Since the volume of data collected is too large for publication here, the subchapters that follow present a simplified summary of the observed data illustrating the radiation situation at the site. The values are rounded and the sensitivity of measurement is not specified. The relatively large span of values observed in some measurements is due to the conditions of measurement at the various sites and during the various seasons and does not represent any increase in the quantity in question during the period covered. Detailed data regarding all the measurements performed, which may be required for an in-depth assessment of the environmental impact of the plant operation, are available from the Temelín plant operator.

2.7.2.4 SPATIAL DOSE EQUIVALENT MONITORING BY THE PARTIAL STATIONS OF THE RADIATION MONITORING SYSTEM IN THE PLANT SURROUNDINGS

The spatial gamma dose equivalent has been monitored by the partial radiation monitoring stations at the following sites: České Budějovice, Bohunice, Zvěrkovice, Nová Ves, Litoradlice, Sedlec, and Týn nad Vltavou, since September 2002 (Týn nad Vltavou: since 2005).

The gamma dose equivalent rates at the above sites are at the level of the natural background.

2.7.2.5 GAMMA PHOTON DOSE EQUIVALENT RATE MONITORING BY USING THERMOLUMINESCENT DOSIMETERS

The mean dose equivalent rate is monitored by using thermoluminescent dosimeters in 3-month intervals. A total of 35 thermoluminescent dosimeters located within the internal emergency planning zone (up to 5 km from the Temelín NPP source) are exposed for 3 months. Included in this number is the TL dosimeter which is located in the area of the Laboratory of Radiation Monitoring of the Temelín NPP surroundings in České Budějovice.

The gamma dose equivalent rates at the above sites are at the level of the natural background. The 2011 data from 35 sites within the internal emergency planned zone plus the Environmental Radiation Monitoring Laboratory in České Buděkovice all the sites monitored are given in 2011. The data obtained from the thermoluminescent dosimeters illustrate the radiation situation at the sites well: in fact, the data for the previous years lie at the same levels, with only minor differences.

Tab. 139 Dose equivalent rates measured inside the Temelín NPP emergency planning zone and at the monitoring site in České Budějovice in 2011

Mean gamma dose equivalent rates measured with themoluminescent dosimeters applying 3-month exposure, 2011 (The combined standard uncertainty of measurement is 16%)									
Site	**nSv/h**				**Site**	**nSv/h**			
	1st Q	**2nd Q**	**3rd Q**	**4th Q**		**1st Q**	**2nd Q**	**3rd Q**	**4th Q**
Neznašov, No. 71	151	162	158	174	Nová Ves, No. 2	118	135	130	143
Bohunice, SRKO	107	120	118	131	Kočín, No. 8	121	145	128	151
Červený Vrch	112	131	130	143	Dříteň, No. 116	123	140	123	141
Týn nad Vltavou,	112	133	128	139	Strachovice,	110	136	130	143
Záluží, U Válků	113	134	132	144	Malešice, No. 36	107	123	114	132
Týn nad Vltavou, Kindergarten	119	133	128	142	Malešice, farmhouse	101	116	109	125
Zvěrkovice, SRKO	117	133	128	141	Sedlec, SRKO	98	116	108	121
Hněvkovice SOU	110	124	115	132	Lhota P.H. No- 27	132	142	144	152
U Palečků	111	125	125	136	Lhota Pod Horami	120	129	125	139
Hněvkovice, dam	116	130	123	141	Lhota P.H. cow house	1114	131	121	140
Litoradlice, No. 11	108	119	116	131	Planovy, No. 38	146	157	153	166
Litoradlice, SRKO	117	134	129	142	Temelín, policlinic	129	142	131	151
Hůrka	107	124	116	133	Temelín, meteorology station	120	133	128	142
Býšov, ČEZ area	106	121	118	132	Všemyslice, No. 36	152	167	158	179
Býšov	105	124	119	133	Všemyslice, No. 33	1118	129	120	140
Coufalka, gamekeeper's lodge	113	125	127	139	NPP area, SRKO	121	129	119	144
Coufalka	113	126	132	138	České Budějovice LRKO	142	147	145	163
Nová Ves, SRKO	130	138	140	149					

2.7.2.6 GAMMA PHOTON DOSE EQUIVALENT RATE MONITORING BY USING PORTABLE INSTRUMENTS

Gamma photon dose equivalent rate measurements are performed in parallel to the in situ measurements at the same sites as where the gamma spectrometric field measurements are made. Portable instruments are used for measurements at 1 m above the ground.

The gamma photon dose equivalent rates are at the level of natural background at all of the monitoring sites, as demonstrated by

Tab. 140 and Tab. 141 below.

Tab. 140 Photon dose equivalent rate ranges measured with portable instruments on uncultivated soils of the villages in the surroundings of the plant, various seasons of 2005 - 2011

Site	nSv/h (approx.)
Nová Ves	50 - 140
Litoradlice	60 - 140
Zvěrkovice	60 - 170
Bohunice	100 - 160
Sedlec	60 - 140

Tab. 141 Photon dose equivalent rate ranges measured with portable instruments on cultivated soils of the villages in the surroundings of the plant, June 2005 - 2011

Site	nSv/h (approx.)
Kočín	120 - 150
Knín	100 - 170
Křtěnov	100 - 140
Temelín, at the orchards	100 - 130

2.7.2.7 FIELD GAMMA SPECTROMETRY

Field gamma spectrometric measurements on uncultivated soil are performed at 5 sites in 3-month intervals. Measurements on cultivated soil are performed at 4 sites once a year. A portable HPGe detector without collimation, positioned 1 m above ground, is used. The presence of artificial radionuclides on the surface only, as the situation would be shortly after fallout, is assumed for the calculation.

Differences between the various sites are commonplace and only document the variability of occurrence of radionuclides at the various sites of the surface of the ground. The areal activities of artificial radionuclides and mass activities of natural radionuclides (^{40}K, U series, Th series) are measured.

From among artificial radionuclides, only ^{137}Cs (arising from the Chernobyl fallout) is measured in cultivated and uncultivated soil in the surroundings of the Temelín NPP. The activities of other artificial radionuclides lie below the lowest detectable levels.

Tab. 142 Areal activities of the artificial radionuclides ^{137}Cs and ^{134}Cs and mass activities of the natural radionuclides ^{40}K and radionuclides in the uranium and thorium series on uncultivated soil, 2005 – 2011

Site	^{137}Cs	^{134}Cs	^{40}K	U series	Th series
	[Bq.m^{-2}]	[Bq.m^{-2}]	[Bq.kg^{-1}]	[Bq.kg^{-1}]	[Bq.kg^{-1}]
Nová Ves	150 - 660	110 - 200	340 – 885	10 - 70	10 - 50
Litoradlice	150 – 1120	150 - 210	340 - 720	10 - 50	10 - 40
Zvěrkovice	380 - 2070	140 - 270	250 - 805	10 - 65	15 - 45
Bohunice	390 - 1420	170 - 270	410 - 650	10 -50	10 - 45
Sedlec	150 - 1020	130 - 250	380 - 615	10 - 40	25 - 40

Tab. 143 Areal activities of the artificial radionuclides ^{137}Cs and ^{134}Cs and mass activities of the natural radionuclides ^{40}K and radionuclides in the uranium and thorium series on cultivated soil, 2005 – 2011

Site	^{137}Cs	^{134}Cs	^{40}K	U series	Th series
	[Bq.m^{-2}]	[Bq.m^{-2}]	[Bq.kg^{-1}]	[Bq.kg^{-1}]	[Bq.kg^{-1}]
Kočín	250 -800	170 - 200	595 - 700	40 - 80	27 -40
Knín	480 - 600	180 - 220	620 - 720	20 - 100	30 - 60
Křtěnov	500 - 1000	170 - 190	630 - 700	30 - 40	30 - 48
Temelín, at the orchards	215 - 510	160 - 180	690 - 750	30 - 40	30 - 40

2.7.2.8 SOIL

For a check, 2 soil layers were sampled by using special sampling equipment, at depths of 0 to 5 cm and 5 to 10 cm under the ground, respectively. The sampling was performed at a known uncultivated soil area at the measuring site. Grain size fraction <2 mm was used for the measurement.

From among the artificial radionuclides, only ^{137}Cs, arising from global fallout, is measurable.

Tab. 144 Activities of radionuclides in soil (per kg dry matter)

Site	Layer	^{134}Cs	^{137}Cs	^{40}K
		[Bq.kg^{-1}]	[Bq.kg^{-1}]	[Bq.kg^{-1}]
Nová Ves	0 to 5 cm	0.4 - 0.6	8 - 21	650 - 820
	5 to 10 cm	0.4 - 0.6	11 - 24	650 - 1000
Litoradlice	0 to 5 cm	0.4 - 0.6	20 - 60	680 - 920
	5 to 10 cm	0.4 - 0.6	10 -20	680 - 770
Bohunice	0 to 5 cm	0.4 - 0.6	18 - 50	670 - 770
	5 to 10 cm	0.4 - 0.6	18 - 45	670 - 760
Sedlec	0 to 5 cm	0.4 - 0.6	9 -30	650 -1000
	5 to 10 cm	0.4 - 0.6	7 - 25	600 - 980

2.7.2.9 MONITORING OF THE MUNICIPAL WASTE DUMP AT TEMELÍNEC

The gamma photon dose equivalent rate at the municipal waste dumping site at is monitored to the extent specified in SÚJB Decision No. 9343/4.3/05. The measurements are conducted in 3 points 1 m above the surface of the dump, once a week.

Tab. 145 Gamma dose equivalent rate levels, 2011

Point 1	Point 2	Point 3	Point 4	Background
[nSv.h^{-1}]	[nSv.h^{-1}]	[nSv.h^{-1}]	[nSv.h^{-1}]	[nSv.h^{-1}]
43 - 93	46 - 88	46 - 82	42 - 74	102 - 157

2.7.2.10 AEROSOLS AND RADIOIODINE GAS

Aerosols for the determination of gamma nuclides are sampled by the partial stations of the radiation monitoring system for the plant surroundings by using large-volume filtering equipment.

Air for the determination of the gaseous species of ^{131}I is sampled in the partial radiation monitoring station at Týn nad Vltavou by using large-volume filtering equipment. The iodine trapping filters are measured by gamma spectrometry after exposure for a week.

From among artificial radionuclides, only ^{7}Be (mainly arising from the action of cosmic radiation) and ^{137}Cs (from Chernobyl fallout) are measured in aerosols in the Temelín NPP surroundings. The activities of other artificial radionuclides lie below the lowest detectable levels. Gaseous ^{131}I levels lie below the MDA as well. Yearly pooled aerosol filter samples also serve to determine volume activity of ^{90}Sr by extraction of ^{90}Y into TBP and measurement based on Cherenkov radiation or, via separation of strontium, by liquid scintillation chromatography on an SrSpec column.

Tab. 146 Typical radionuclide activities in a pooled aerosol sample from the surroundings of the Temelín NPP

^{7}Be	^{137}Cs	^{40}K	^{210}Pb
[μBq.m^{-3}]	**[μBq.m^{-3}]**	**[μBq.m^{-3}]**	**[μBq.m^{-3}]**
approx. 770 - 8000	approx. 1 - 7	approx. 20 - 140	approx. 120 - 1200

Tab. 147 Activity of gaseous radioiodine I-131 measured by SRKO at Týn nad Vltavou

Year	I-131 [μBq.m^{-3}]
2007	**approx. 120 - 340**
2008	**approx. 120 - 300**
2009	**approx. 110 - 380**
2010	**approx. 260 - 450**
2011	**approx. 270 - 430** **Max 684 (6 April 2011, affected by the Fukushima accident)**

2.7.2.11 ATMOSPHERIC FALLOUT

Atmospheric fallout is sampled by means of large-area sampling equipment located at the partial radiation monitoring stations at Litoradlice and Zvěrkovice (at sites with prevailing wind direction). The sampling area is 0.5 m^{2}. Wet and dry fallout cannot be distinguished. The fallout is evaluated once a month. After acidification, the samples are processed by evaporation without preceding filtration. Gamma nuclides are determined by gamma spectrometry using a HPGe detector at 30% or 40% relative efficiency. Only the ^{7}Be and ^{137}Cs radionuclides, arising from global fallout, are measured.

Tab. 148 Areal activity of ^{7}Be and ^{137}Cs from atmospheric fallout, 2007 - 2011

Site	^{7}Be [Bq.m^{-2}]	^{137}Cs [Bq.m^{-2}]
Litoradlice	approx. 3 - 70	approx. 0.2 - 0.3
Zvěrkovice	approx. 6 - 135	approx. 0.2 - 0.4

2.7.2.12 RAINWATER

Rainwater is sampled continuously by a stationary sampling system at the Temelín meteorological station. The radionuclide of interest in rainwater is tritium. The samples are distilled and their beta activity is measured on a liquid scintillation spectrometer.

Measurements over the 2007 – 2011 period show that the volume activity of tritium has been constantly below 3 Bq/l.

2.7.2.13 SURFACE WATERS

Surface water is sampled in 10 litre volumes for the determination of gamma nuclides. The samples are preconcentrated by evaporation and transferred to a 1000 ml Marinelli vessel for gamma spectrometric analysis. Total alpha and beta activities are determined by using samples in 3 litre volumes.

Samples for the determination of tritium are taken separately in 250 ml volumes, distilled, and their beta activity is measured on a liquid scintillation spectrometer.

Once a year, surface waters are also analysed for the volume activity of ^{90}Sr by extraction of ^{90}Y into TBP and measurement based on Cherenkov radiation or, via separation of strontium, by liquid scintillation chromatography on an SrSpec column.

The volume activities in surface waters within the Temelín NPP area and in its surroundings (Vltava - Solenice, Hněvkovice, Hladná and Bělohůrecký pond) have been mainly at the levels shown below during the past years:

^{137}Cs approx. 0.01 Bq/l,

^{3}H approx. 3 – 30 Bq/l,

Total α approx. 0.01 – 0.04 Bq/l,

Total β approx. 0.05 – 0.2 Bq/l,

Higher levels are occasionally observed, especially in Solenice and Hladná, which lie downstream of the site of discharge from the NPP. For instance, a tritium volume activity level of 621 Bq/l was measured at Hladná on 10 May 2012. Nevertheless, even this extreme is deeply below the water pollution limit of 3500 Bq/l laid down by Government Decree No. 61/2003 Coll. [L. 286].

2.7.2.14 SEDIMENTS

The surface sediment layer is sampled at the surface water sampling sites once a year. The samples are taken to a depth of 5 cm by means of a small spade or a special sampling device. The fraction <2 mm is used for the measurement.

From among the artificial radionuclides, only ^{137}Cs, arising from global fallout, is measurable. Specific activities up to approx. 40 Bq/kg are measured.

2.7.2.15 GROUNDWATER

Groundwater data from the area of the Temelín nuclear power plant have been collected and compared since 2000 within the "Monitoring and Evaluation Programme for the Environmental Impacts of the Temelín Nuclear Power Plant". The annual data update and evaluation also concerns groundwater quality. Parameters measured include, among others, total volume alpha activity, potassium 40, volume activity of cesium 137, and tritium 3.

The groundwater samples are subjected to gamma spectrometric analysis for the determination of gamma nuclides.

Tritium is measured by beta liquid scintillation spectrometry.

Volume activities observed in groundwater at the Temelín NPP site and in its surroundings in 2011 were as follows:

^{137}Cs	<0.02 Bq/l
^{3}H	<3 Bq/l.

2.7.2.16 DRINKING WATER

Drinking water is sampled in 10 litre volumes for the determination of gamma nuclides. The samples are preconcentrated by evaporation and transferred to a 1000 ml Marinelli vessel for gamma spectrometric analysis.

Two-litre samples are taken for the determination of total alpha and beta volume activity. Radiochemical processing and analysis comply with ČSN 757611 and ČSN 757612. Samples for the determination of tritium are taken separately in 250 ml volumes, distilled, and their beta activity is measured on a liquid scintillation spectrometer.

Volume activities of tritium (^{3}H) in water taken from the water mains in Dříteň and Týn nad Vltavou and from wells in Křtěnov, Kočín and Temelín were below approx. 3 Bq/l in 2011.

2.7.2.17 MILK

Milk is sampled once in 2 weeks. The samples (3000 ml) in the native state are measured with a semiconductor detector at a relative efficiency of 30% or 40%.

Pooled yearly milk samples are analysed for the volume activity of ^{90}Sr by extraction of ^{90}Y into TBP and measurement based on Cherenkov radiation or, via separation of strontium, by liquid scintillation chromatography on an SrSpec column.

From among the artificial radionuclides, only ^{137}Cs, arising from global fallout, is measurable in milk. The activities of other artificial radionuclides lie below the lowest detectable levels.

Milk from the Dynín Farm Cooperative (Bohunice cow house) exhibited ^{137}Cs volume activities lower than 0.10 to 0.13 Bq/l in 2011.

2.7.2.18 AGRICULTURAL AND FOREST CROPS AND FRUITS

Samples of agricultural and forest crops and fruits are taken in dependence on the growing season. A pooled sample from several points of each plot, representing the average of the plot, is used. Samples from suppliers are accepted in the native state.

Samples of agricultural produce and similar products are taken within a radius of 5 km from the Temelín NPP (with special regard to the protective zone). The sampling sites are selected operatively taking into account the local situation and the sowing plan.

From among the artificial radionuclides, only ^{137}Cs, arising from global fallout, is measurable.

The following cesium and potassium activity levels were measured in cereals and fodder crops grown in the surrounding villages Temelín, Bohunice, Březí and Křtěnov in 2011:

^{137}Cs <2 – 4 Bq/kg

^{40}K <130 – 930 Bq/kg.

In apples and in bilberries, mass activity of ^{137}Cs was below 0.9 Bq/kg and 1.4 Bq/kg, respectively.

2.7.2.19 FISH AND GAME

Based on the monitoring programme, fish samples are taken once a year from the Orlík reservoir. Gamma radiation is measured in fish muscles in the native state. From among the artificial radionuclides, only ^{137}Cs, arising from global fallout, is measurable.

^{137}Cs mass activities in the fish lay within the range of 0.3 – 0.7 Bq/kg in 2011 [L. 42]. Boar meat exhibited ^{137}Cs at approx. 20 Bq/kg.

2.7.2.20 DISPERSION AND DILUTION IN SURFACE WATERS

Liquid discharges from the Temelín NPP systems where water may be contaminated with radionuclides are collected in monitoring basins and are pumped into the drain channel only after radiochemical analysis which confirms that the permitted limits are not exceeded. The final inspection of the industrial wastewaters is at the outlet from the plant area, in building SO 362/02.

The degree of dispersion and dilution of the discharges into surface waters depends of the dynamics of the Vltava cascade, which can be divided into 3 segments from this point of view: Orlík reservoir, Kamýk-Slapy reservoir system, and Štěchovice reservoir - Prague-Podolí profile (with significant offtake for water use purposes). Available information on the dilution and holdup time in the segments can be found, e.g., in Section 2.4.12 of the Temelín-1,2 pre-operation safety report [L. 18] and in report [L. 246].

The hydrological surface measurement programme is a part of ČEZ's Temelín NPP surroundings monitoring programme. The extent, frequency and method of monitoring in the Vltava, local watercourses, sediments and fish during the normal radiation situation and in surface waters during emergency associated with potential radiological impacts are laid down in ČEZ methodology ME 0203 [P. 48]. Outputs from that Temelín NPP surroundings monitoring programme include records of current specific and volume activities, which will serve as reference levels after the ETE3,4 units are put into operation.

2.7.3 REQUIREMENTS AND CRITERIA

The requirements and criteria for site assessment from the radiological aspect follow from Act No. 18/1997 Coll. [L. 2] and the associated implementing Decree No. 215/997 Coll. [L. 1] as well as from the standard IAEA NS-R-3 [L. 6], as summarised in Tab. 149.

Tab. 149 Requirements and criteria regarding the radiation situation at the Temelín NPP site

ID	Art.	Criterial requirement in Decree No. 215/1997	Paragr.	Requirement in IAEA NS-R-3
8.1	4 a)	Expected exceeding of specified mean annual effective radiation doses to individuals[100] in the critical population group present at the site corresponding to the expected location during operation of the nuclear installation or workplace with a very significant ionising radiation source (henceforth "installation or workplace"),	2.4	Site characteristics that may affect the safety of the nuclear installation shall be investigated and assessed. Characteristics of the natural environment in the region that may be affected by potential radiological impacts in operational states and accident conditions shall be investigated. All these characteristics shall be observed and monitored throughout the lifetime of the installation.
8.2			4.5	A programme of investigation and measurements of the surface hydrology shall be carried out to determine to the extent necessary the dilution and dispersion characteristics for water bodies, the reconcentration ability of sediments and biota, and the determination of transfer mechanisms of radionuclides in the hydrosphere and of exposure pathways.
8.3			4.15.	AMBIENT RADIOACTIVITY Before commissioning of the nuclear installation the ambient radioactivity of the atmosphere, hydrosphere, lithosphere and biota in the region shall be assessed so as to be able to determine the effects of the installation. The data obtained are intended for use as a baseline in future investigations.

[100] State Office for Nuclear Safety Regulation No. 307/2002 Coll. [L. 4], on radiation protection, which replaced Regulation No. 184/1997 Coll., referred to in Regulation No. 215/1997 Coll. [L. 1].

ID	Art.	Criterial requirement in Decree No. 215/1997	Paragr.	Requirement in IAEA NS-R-3
8.4			4.3	On the basis of the data obtained from the investigation of the region, the atmospheric dispersion of radioactive material released shall be assessed with the use of appropriate models. These models shall include all significant site specific and regional topographic features and characteristics of the installation that may affect atmospheric dispersion.
8.5			4.6	An assessment of the potential impact of the contamination of surface water on the population shall be performed by using the collected data and information in a suitable model.

2.7.4 DOCUMENTS PROVIDING A BASIS FOR THE ASSESSMENT

- Act No. 18/1997 Coll. on peaceful uses of nuclear energy and ionising radiation (Atomic Act) and on the amendment of some acts [L. 2]
- Decree No. 215/1997 Coll. on criteria for the siting of nuclear installations and very significant ionising radiation sources [L. 1]
- Decree No. 307/2002 Coll. on radiation protection [L. 4]
- Decree No. 319/2002 Coll. on the performance and management of the national radiation monitoring network [L. 25]
- SÚJB BN-JB-1.14, Interpretation of criteria for the siting of nuclear installations and a proposal for evidence documentation, SÚJB, April 2012 [L. 268]
- IAEA NS-R-3 Site Evaluation for Nuclear Instalation, Safety Requirements, Vienna 2003 [**L. 6**]
- IAEA RS-G-1.8 Environmental and Source Monitoring for Purposes of Radiation Protection, Safety Guide, Vienna, 2005 [L. 24]
- IAEA GS-G-4.1: Format and Content of the Safety Analysis Report for Nuclear Power Plants, safety guide No. GS-G-4.1, IAEA, 2004 [L. 272]
- Results of monitoring of discharges from the Temelín nuclear power plant 2005, document ETE/V5020200/5/2005 [L. 36]
- Results of monitoring of discharges from the Temelín nuclear power plant 2006, document ETE/V5020200/5/2006 [L. 37]
- Results of monitoring of discharges from the Temelín nuclear power plant 2007, document ETE/905002240/5/2007 [L. 38]
- Results of monitoring of discharges from the Temelín nuclear power plant 2008, document ETE/905002240/5/2008 [L. 39]

- Results of monitoring of discharges from the Temelín nuclear power plant 2009, document ETE/905002240/5/2009 [L. 40]
- Results of monitoring of discharges from the Temelín nuclear power plant 2010, document ETE/905002240/5/2010 [L. 41]
- Results of monitoring of discharges from the Temelín nuclear power plant 2011, document ETE/905002240/5/2011 [L. 42]
- Preliminary assessment of the severity of the radiation impacts of a severe accident with respect to potential contamination of surface waters, ÚJV Řež, ENERGOPROJEKT PRAHA Division, VÚV TGM Praha, 6/2011 [L. 246]
- Preoperational safety report for Temelín ETE1,2 Units [L. 18]
- ČEZ_ME_0203 - Monitoring programme for the surroundings of the Temelín NPP [P. 48]
- ČEZ_ME_0455 - Monitoring programme for discharges from the Temelín nuclear power plant [P. 49]

2.7.5 METHODS APPLIED TO THE EVALUATION

Evaluation of the radiation situation at the site is based on general principles, which are also included in the recommendation for the method of development and for the extent of the safety-related documentation [L. 272], Sections 3.35 and 3.36:

- The environment at the site must be described with respect to the radiation situation, including the impacts of the nearest nuclear installations and other external sources, to a detail sufficiently describing the initial state and enabling the radiological conditions to be assessed.
- The monitoring system and the associated technical means for the detection of ionising radiation and contamination that are currently used must be briefly described.

The continuous or periodic monitoring and assessment are in accordance with the IAEA recommendation IAEA RS-G-1.8 [L. 24]. The assessment is based on the results of the monitoring system, whose layout is shown in Fig. 60. The impact on an individual in the critical population group is examined so that all exposure pathways are taken into account. Radionuclide migration and distribution in the environment are shown in Fig. 61. This methodology enables the radiation burden to the critical population groups to be expressed in a way that is well suited to the assessment of the acceptability of additional nuclear installations at the site.

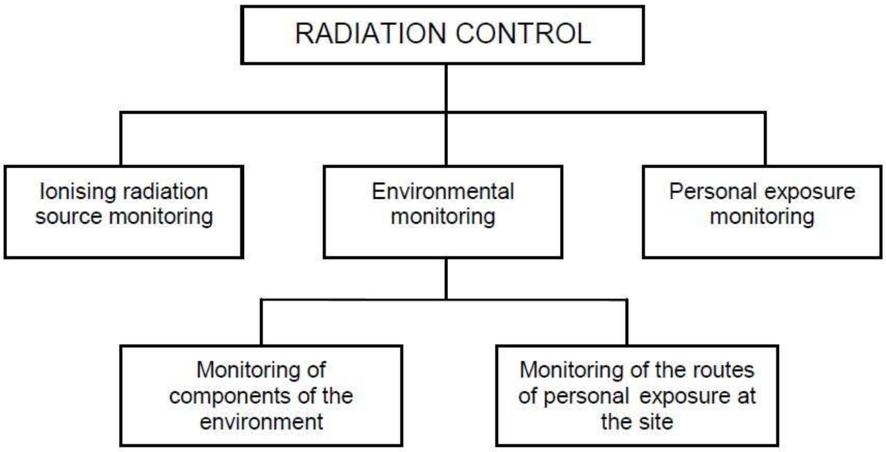

Fig. 60 Extent of monitoring required for the purposes of radiation protection of the population [L. 24]

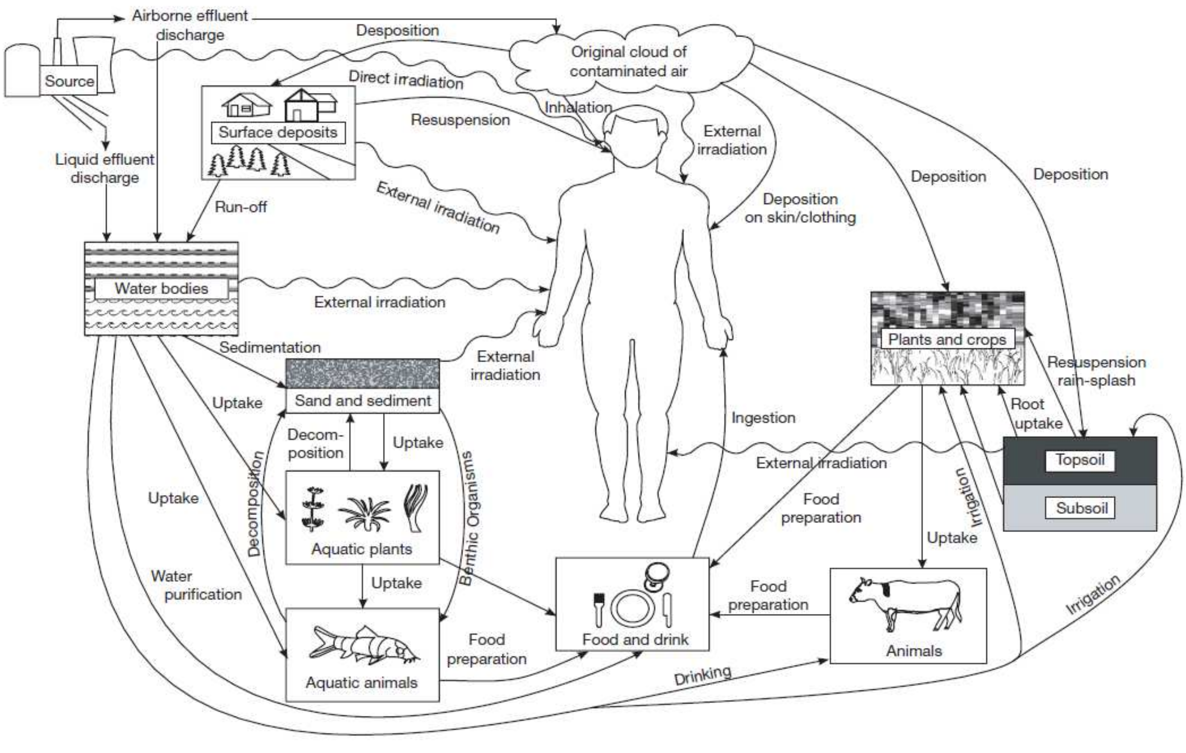

Fig. 61 Potential population exposure pathways arising from radioactivity discharges into the environment [L. 24]

Neither Czech legislation nor IAEA standards lay down specific numerical levels or criteria based on which the suitability of a site for siting a nuclear installation could be assessed from the radiation situation determined by the natural ionising radiation sources. Whether a nuclear installation can be sited or not is basically not determined by the site's natural properties. In fact, what plays a role is the impact of existing nuclear installations if any and of the technological solution of the nuclear installation in question if the principles of a reasonably achievable radiation protection level are not respected. A case where the dose optimisation limit laid down by Article 56 paragraph 3 of Decree No. 307/2002 Coll. [L. 4] would be exceeded in a planned exposure situation, i.e. where individuals in the critical group would receive an effective dose from atmospheric discharges >200 µSv and an effective dose from liquid discharges >50 µSv during a calendar year, constitutes an exclusion criterion.

2.7.6 DEFINITION OF THE AREA EXAMINED

Decree No. 215/1997 Coll., [L. 1] Article 2 defines the locality as an area up to 20 km distance from the boundary of land proposed for the siting, and the site vicinity zone as an area up to 3 km distance from the boundary of land proposed for siting. This definition is also adequate from the radiation situation assessment point of view because the area includes villages with critical population groups and is fully covered by the monitoring programme which is in place.

2.7.7 DETAILED ASSESSMENT OF ALL REQUIREMENTS AND CRITERIA SPECIFIED BY DECREE NO. 215/1997 COLL. IN COMBINATION WITH THE IAEA NS-R-3 STANDARD

2.7.7.1 CRITERION DEFINED BY ARTICLE 4A) OF DECREE NO. 215/1997 COLL.

The text of the criterion specified in Article 4 paragraph a) of Decree No. 215/1997 Coll. [L. 1] is reproduced in Tab. 149 under item 8.1.

The siting of additional reactor units at the Temelín NPP site would not be permissible if the current ETE1,2 units contributed so much to the effective doses that it would be reasonable to expect that the optimisation limits for the introduction of radionuclides into the environment would be exceeded after making the additional units operable.

Actually, however, the results of assessment of the radiation situation reported in Section 2.7.2 give evidence that the effect of operation of the current ETE1,2 units on an individual in the critical population group is insignificant: both in the critical group that is most exposed to discharges into air and in the critical group that is most exposed to discharges into watercourses the effective doses / dose commitments to individuals in the various age groups are lower than 1 µSv.

According to generally accepted optimisation principles, a reasonably achievable radiation protection level is demonstrated if the annual effective dose does not exceed 50 µSv. It will be clear from a comparison of this optimisation exposure level and the current radiation burden of the affected population that the conditions at the site permit installation of any new nuclear installation whose radioactivity discharge design respects the principle of a reasonably achievable level of radiation protection.

This is borne out by the expected volume of radioactive discharges from the ETE3,4 units specified in the preliminary assessment of the impacts on the employees, population and the environment – refer to Chapter 4 of the reference safety report.

In conclusion, the limiting mean annual effective doses received by individuals, including individuals in the critical group, as laid down by applicable legislation (Decree No. 307/2002 Coll. [L. 4) will not be exceeded. Furthermore, it is reasonable to expect that prior to launching, the proposed authorised limits will be below the level of 50 µSv of the annual effective doses to individuals in the most impacted population groups.

2.7.7.2 REQUIREMENT IN SECTION 4.5 OF IAEA NS-R-3

The text of the requirement in Article 4.5 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 149 under item 8.2.

A monitoring programme has been developed in relation to the operation of the existing ETE1,2 units. This programme includes, among others, measurement of surface waters, sediments, groundwater and sources of drinking water. The measurements are the responsibility of the Radiation Monitoring Laboratory for the NPP site surroundings. The results are published periodically – see, e.g., [L. 36] to [L. 42].

Furthermore, the dilution of discharges and the holdup times in the various segments of the Vltava cascade have been mapped, as described in Section 2.7.2.20 above.

In summary, the programme demanded was developed earlier in relation to the preparation for operation and with the operation of ETE1,2 units and the measurements have been performed for a number years. Practical results give evidence that the dilution ratios are satisfactory and no adverse radionuclide accumulation impacting the biota takes place. Since the same discharge point is planned for the new units, the empirically verified assumptions are applicable to the impact assessment of the new nuclear sources as well.

2.7.7.3 REQUIREMENT IN SECTION 4.15 OF IAEA NS-R-3

The text of the requirement in Article 4.15 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 149 under item 8.3.

The radiation situation in the environment affected has been monitored thoroughly in a documented manner. (The results for the 2005-2011 period are summarised in reports [L. 36] to [L. 42]). All the compartments of the environment continue to be monitored in accordance with approved monitoring programmes.

The continuously acquired and stored monitoring results will constitute a suitable source of information, allowing the environmental impacts of the new nuclear sources to be evaluated once the reactor units are made operable.

2.7.7.4 REQUIREMENTS IN SECTION 4.3 OF IAEA NS-R-3

The text of the requirement in Article 4.3 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 149 under item 8.4.

The NORMAL ver. 02 software employed uses a model of radionuclide propagation in air which was developed based on the specific topographic features of the landscape. Parameters describing the meteorological conditions as derived from the measurements were also used in the calculations. The amount of radioactive substances discharged into air was estimated conservatively by using information acquired from the operation of the existing nuclear installations. Combined with the conservative model of radioactivity propagation in air, this estimate provides the maximum theoretical committed dose received by a representative individual.

2.7.7.5 REQUIREMENTS IN SECTION 4.6 OF IAEA NS-R-3

The text of the requirement in Article 4.6 of IAEA NS-R-3 [L. 6] is reproduced in Tab. 149 under item 8.5.

Radioactivity propagation in surface waters was assessed by using a conservative model (approved by the SÚJB) for estimating the potential contribution of contamination of wastewaters arising from the operation of the existing units ETE1,2 to the committed dose received by a representative individual.

2.7.8 CONCLUDING ASSESSMENT

The following conclusions can be made based on the analyses and assessments of the radiation situation at the site of the ETE3,4 units in accordance with applicable criteria and requirements.

- Operation of the future ETE3,4 units will not result in mean annual effective radiation doses received by individuals in the critical population group exceeding the regulatory limits, and the criterion specified in Article 4 paragraph a) of Decree No. 215/1997 Coll. [L. 1] will be complied with. The radiation situation within the wide area, due to natural influences combined with the contribution from the operation of the ETE1,2 units, is at a level which is normal within the whole Czech Republic. The degree of environmental contamination with radionuclides also lies within the range which is normal within the country and matches the decreasing trend of historical fallout since 1986, and does not burden public health to an extent requiring any corrective or regulatory measures.
- Information regarding initial radiation at the ETE3,4 site was analysed in accordance with the requirements specified in Sections 2.4, 4.5, and 4.15 of the IAEA NS-R-3 standard [L. 6]. Systematic radiation situation monitoring at the site provides the requisite background information for a subsequent evaluation of the environmental impacts of the operation of the new reactor units.

The analyses presented in Section 2.7 did not point to the necessity for any design provisions or project basis.

These facts warrant the conclusion that the radiation situation does not preclude in any way the siting of a new nuclear installation.

2.8 EMERGENCY PREPAREDNESS AT THE SITE

2.8.1 SCOPE OF THIS SECTION

This section assesses the suitability of the Temelín site for the siting of a new nuclear source of the PWR type Generation III or III+ providing electricity within the planned output range, now with respect to the potential necessity to expand the existing boundaries of the ETE1,2 emergency planning zone.

The objective of emergency preparedness is to ensure that the ionising radiation doses received by the population in the event of a radiation accident do not exceed acceptable levels. This implies that the conditions at the site must enable urgent protective measures to be promptly introduced to the full extent. The criteria applied to the site assessment procedure with respect to emergency preparedness are based on Article 4 paragraph 5 of Act No. 18/1997 Coll. [L. 2] and decrees implementing that act, in particular Article 4 paragraph b) of Decree No. 215/1997 Coll. [L. 1] and Articles 98-100 of Decree No. 307/2002 Coll. [L. 4]. The requirements following from Section 2.27 of IAEA NS-R-3 [L. 6], ICRP Recommendation 103 as regards exposures during accidents [L. 254], Decree No. 318/2002 Coll. [L. 26], and from facts published in the technical document IAEA TECDOC-953 [L. 23] were applied as well.

2.8.2 SUMMARY OF EXISTING FACTS REGARDING EMERGENCY PREPAREDNESS

2.8.2.1 BACKGROUND INFORMATION

The assessment of the conditions at the site for the siting of the ETE3,4 units with respect to emergency preparedness is based on 2 major sources of information:

- Information regarding potential environmental impacts of potential accidents at the ETE3,4 units, and
- Information regarding emergency preparedness at the existing Temelín NPP

2.8.2.2 SUMMARY OF KNOWLEDGE CONCERNING POTENTIAL IMPACTS OF RADIATION ACCIDENTS AT THE ETE3,4 UNITS

When selecting the design solution for the ETE3,4 units, stress is laid, among other factors, on a reduction of the likelihood of a radiation accident and on minimisation of the scope of protective measures required if a radiation accident occurs. When formulating requirements for reducing the impacts of a radiation accident on the surroundings, an incremental approach was adopted, where the demands for minimisation of the impacts on the population increase with increasing probability of accident (See Chapters 3 & 4 of the reference safety report).

2.8.2.3 SUMMARY OF EXISTING FACTS REGARDING EMERGENCY PREPAREDNESS AT THE SITE

The area of the emergency preparedness zone at the Temelín site was defined by SÚJB Decision No. 311/1997 of 5 August 1997 [L. 28] based on a proposal submitted by ČEZ a.s. as the licence holder (refer to Drawing Dwg. 1 in the annex hereto).

Site analyses regarding radioactivity propagation in the event of radiation accident were developed as a tool for definition of the area of the emergency preparedness zone. The results were used to devise personnel and population protection measures.

The internal part of the zone, with a radius of approx. 5 km from the existing ETE1,2 units encompasses areas of 5 villages/towns: Dříteň, Olešník, Temelín, Týn nad Vltavou and Všemyslice (for a list of the towns and villages and their associated parts see Tab. 15). Parts of the territories of the towns/villages except for the village of Temelín lie in the outer area of the emergency planning zone.

This outer area, 5 to 13 km from the plant site (ETE1,2), is divided into 16 sectors. Towns/villages falling in this area are listed in Tab. 16.

Apart from the emergency planning zone, a protective zone is defined around the building site, and restrictive conditions for activities within this zone are specified. From the emergency preparedness aspect, it is important that permanent inhabitation of this zone is not permitted [L. 237] [L. 243]. Activities within the protective zone are also limited by the fact that the licence holder is the owner of land both inside the plant area and in its nearest surroundings (see list of land owned by ČEZ a. s. as of February 2011 [L. 244])

The external emergency plan includes the following provisions to reduce exposure of people and the environment during radiation accidents [L. 20]:

- Urgent protective measures including sheltering, iodine prophylaxis and evacuation
- Subsequent protective measures including relocation, control of the consumption of food and water contaminated by radionuclides, and control of the use of feedstuffs contaminated by radionuclides.

Protective measures will be taken if their benefit outweighs their cost and the damage they cause, and they should be optimised in terms of form, scope and duration so as to provide as much benefit as reasonably achievable.

The reference levels of the appropriate intervention level specified in the plant's internal emergency plan will serve as the basic guidance to those taking decisions regarding the introduction of protective measures.

Information describing the residential patterns and infrastructure in the surroundings of the NPP, which are factors relevant to the expected collective effective dose and to the feasibility of implementation of the protective measures, were taken into account when defining the intervention levels. These include but are not limited to the following:

- Presence of specific population groups (hospitals, homes for the elderly, nursing homes, ...)
- Traffic situation
- Population density and major towns/villages

Considerations to be made when taking a decision on protective measures in an radiation emergency situation include, in particular, the question as to whether the current situation differs appreciably from the conditions considered when defining the intervention levels, and whether another source of danger to the population exists in parallel.

Emergency preparedness consists of 2 phases: the planning phase and the intervention phase.

The planning phase includes and is implemented though:

a) Technical provisions:
   - Establishment and maintenance of supporting emergency centres, shelters, assembly points
   - Establishment and maintenance of a technical warning system for people within the NPP area and protective zone
   - Establishment and maintenance of a technical system for communicating information about emergencies
   - Establishment and maintenance of a technical warning system for population within the emergency planning zone
   - Ensuring transport means (evacuation coaches, emergency vehicles)

b) Personnel provisions:
   - Establishment of a system or organising response to an emergency – definition of responsibilities/positions
   - Appointment of specific persons to specific positions within the emergency response organisation
   - Training of all employees, including training of suppliers' employees organised as part of the site admittance procedure. (The scope of the training is defined in the plan of theoretical and practical training, which includes emergency preparedness issues.)
   - No admittance of visitors to the plant area unless accompanied by an authorised manager or employee.

c) Organisational provisions:

- Preparation and period update of the Internal Emergency Plan
- Creation of an event classification system
- Development of documents such as directives and guidelines, procedures, shared documents, methodologies, intervention instructions
- Specification of responsibilities of the positions in the emergency response organisation
- Creation and maintenance of a system of readiness within the emergency response organisation
- Purchase and periodic replacement of antidotes for persons present within the NPP and for population within the emergency planning zone
- Definition of requirements and conditions for suppliers
- Transfer of documents required for the development of the external emergency plan
- Preparation and distribution of information materials for the population within the emergency planning zone
- Purchase and periodic replacement of emergency protective equipment for ČEZ employees at the Temelín NPP site and within the protective zone.

d) Emergency preparedness testing:

- Implementation of planned emergency exercises
- Testing of the technical equipment
- Documented inspection of the equipment
- Recording and storage of records from emergencies
- Final assessment of each exercise and preparation of a record of the assessment which is stored for the next 5 years

The intervention phase includes and is implemented though:

a) Tools to identify an emergency:

- Monitoring of technological nuclear safety and radiation protection related parameters and, in particular, data from the monitoring network operated by the licensee within the scope laid down by Government Decree No. 11/1999 Coll. on the emergency planning zone [L. 283], by the monitoring programme pursuant to Decree No. 307/2002 Coll. [L. 4], and by the plant's internal emergency plan approved by the SÚJB.

b) Emergency severity assessment

c) Announcement of emergency:

- Warning of all employees and any other persons within the plant area and protective zone
- Notification of authorities and organisations
- Warning of population within the emergency planning zone

- Activation of intervention personnel

d) Intervention management and implementation

- Activities of the positions of the emergency response organisation in the management of interventions as specified by the intervention instructions

e) Procedures to reduce exposure of personnel and other persons

- Announcement of protective provisions (sheltering, assembling, prophylaxis, evacuation)
- Recording and monitoring of the movement of individuals within the area
- Use of protective equipment for employees

f) Documenting of activities during emergencies

- Development of a record of emergency and its transfer to the SÚJB

If the ETE3,4 units are constructed, emergency preparedness documents for the ETE3,4 units must be prepared and emergency preparedness documents for the ETE1,2 units must be updated for the period of construction. After all the 4 units are operable, documentation for NPP units 1, 2, 3 and 4 must be developed as follows:

- The size of the emergency planning zone area must be re-evaluated
- During the construction phase, provisions (both organisational and material) to include protection of personnel involved in the construction work must be made (expansion of the personnel warning system so as to cover construction workers as well; protective equipment, evacuation means; assembly points; training and exercises).
- A new area must be found to replace Assembly Point C, on which construction work of ETE3,4 will be performed.
- Documents must be available for updating the external emergency plan (e.g. updating the number of population within the emergency planning zone with respect to the expected increase by the construction personnel and other new personnel [plus their families] who will be accommodated within the zone) and for updating the size of the emergency planning zone for NPP units 1, 2, 3 and 4. The evacuation routes planned in the external emergency plan are not expected to be changed by the construction of the new ETE3,4 units.
- An updated internal emergency plan of the plant (taking into account the organisational approach to the interrelation between ETE1,2 and ETE3,4) will be submitted for approval before unit 3 is put in operation. This will address, in particular, the following issues:
    - Interrelation between ETE1,2 and ETE3,4 in the organisation of the emergency response
    - Shelters and assembling points for ETE1,2 and ETE3,4
    - Planned technical means in support of the protective measures

2.8.3 REQUIREMENTS AND CRITERIA FOR EMERGENCY PREPAREDNESS ASSESSMENT

Requirements and criteria for emergency preparedness are the subject of the exclusion criterion in Decree No. 215/1997 Coll. [L. 1]. Requirements for protection of the population are also laid down in Act No. 18/1997 Coll. [L. 2].

Tab. 150 Requirements and criteria for emergency preparedness assessment (Decree No. 215/1997 Coll.)

ID	Art.	Criteria requirement in Decree No. 215/1997	Par.	Requirement in IAEA NS-R-3
9.1	4b)	Unfeasibility of a timely introduction and complete implementation of all urgent provisions to protect the population 1) during a radiation accident of an installation or workplace, especially because of the population distribution and existence of residential formations at the site where the siting is planned		No requirements for emergency planning are specified in Sections 3, 4 or 5.

2.8.4 DOCUMENTS AND INFORMATION FOR EMERGENCY PREPAREDNESS ASSESSMENT

- Act No. 18/1997 Coll. on peaceful uses of nuclear energy and ionising radiation (Atomic Act) and on the amendment of some acts (as amended) [L. 2]
- Decree No. 423/2001 Coll. specifying the procedure and scope of assessment of natural medicinal sources and natural mineral water sources and other details of their use, requirements for the environment and equipment of spas and requirements for expert opinions regarding the usability of natural medicinal sources and climatic conditions for healing purposes, natural mineral water for the manufacture of packed natural mineral waters, and on the condition of the environment of natural medicinal spas (Regulation on Sources and Spas) (as amended [L. 3])
- Decree No. 318/2002 Coll. on details of emergency preparedness of nuclear installations and workplaces with ionising radiation sources and on requirements for the content of the internal emergency plan and emergency rules (as amended) [L. 26]
- IAEA NS-R-3 Site Evaluation for Nuclear Installations, Safety Requirements, Vienna, 2003 [L. 6]
- IAEA GS-R-2 Preparedness and Response for a Nuclear or Radiological Emergency. Requirements, Vienna, 2002 [L. 21]
- IAEA GS-G-2.1 Arrangements for Preparedness for a Nuclear or Radiological Emergency, Vienna, 2007 [L. 22]
- IAEA TECDOC-953 Method for Developing Arrangements for Response to a Nuclear or Radiological Emergency, Technical documents. Vienna, 2003 [L. 23]

- SÚJB BN-JB-1.14, Interpretation of criteria for the siting of nuclear installations and a proposal for evidence documentation, SÚJB, April 2012 [L. 268]
- External emergency plan of the Temelín nuclear power plant, June 2006 [L. 20]
- Regulation of the District Government in České Budějovice on the promulgation of a protective zone of the Temelín nuclear power plant, 26 September 1985 [L. 243]
- Decision No. 311/1997, ref. 4715/4.0/97/Prz, of 5 August 1997 on the area of the Temelín NPP emergency planning zone [L. 28]
- Government Decree No. 11/1999 Coll. on emergency planning zone [L. 283]

2.8.5 METHODS APPLIED TO THE EVALUATION

Requirements for the methodology of site assessment with respect to the introduction of emergency plans aimed at reducing committed doses received by the population in an emergency situation are laid down in Article 5 and Article 19 of Act No. 18/1997 Coll. [L. 2], Article 4 paragraph b) of Decree No. 215/1997 Coll. [L. 1], paragraphs 2.1c), 2.2.7, and 2.29 of IAEA NS-R-3 [L. 6], paragraphs 3.17, 4.48a), 4.50, 4.51, 4.67, and 4.91 of IAEA GS-R-2 [L. 21], and Annexes V, VI, and VII of IAEA GS-G-2.1 [L. 22].

The site suitability assessment also took into account safety objectives described in the EUR document [L. 264], which is a basic document for formulating technical and safety requirements for the supplier of the new nuclear installation. In accordance with those safety objectives, the ETE3,4 units will be designed so that no urgent protective measures need to be introduced at distances larger than 800 m and no follow-up protective measures need to be introduced at distances larger than 3 km.

Since the planned building site lies within the existing protective zone with no permanent inhabitants, the assessment also included the reach of radiation impacts with respect to the existing protective zone boundary.

2.8.6 DEFINITION OF THE AREA EXAMINED

The area under assessment matches the emergency planning zone of ETE1,2, defined in documentation [L. 20]. This is an area with a radius of approx. 13 km whose boundaries were defined taking into account the shape of the ground and objects present. The zone is drawn in the map reproduced in the Annex hereto [Dwg. 1].

2.8.7 DETAILED ASSESSMENT OF ALL REQUIREMENTS AND CRITERIA DEFINED IN DECREE NO. 215/1997 COLL., IN COMBINATION WITH IAEA NS-R-3

2.8.7.1 CRITERION ACCORDING TO ARTICLE 4 PARAGRAPH 4b) OF DECREE NO. 215/1997 COLL.

The text of the criterion defined in Article 4 paragraph b) of Decree No. 215/1997 Coll. [L. 1] is reproduced in Item 9.1 in Tab. 150.

An emergency plan for the existing nuclear installations, i.e. the Temelín NPP (ETE1,2, each with a VVER1000 reactor) and a spent fuel storage facility with a capacity of 1370 t at the Temelín site, exists, has been approved and is functional. The plant extension with 2 additional PWR units with electric output up to 3,400 MW will not appreciably change the requirements for:

- the emergency planning zone and for the protective measures within this zone in case of radiation accident
- population distribution, village and house arrangement patterns
- throughput (capacity) of roads

The change in the number of NPP employees from the current 1200 to the expected 2000 will not result in an exceeded capacity of the Class II roads which connect the plant to the country's road network.

Prior to the start of the construction work, additional areas that would serve as the employees' assembling point (e.g. for evacuation) will be sought to replace the current assembling point which lies at the future NPP3, 4 building site. The necessity of constructing additional shelters within the plant area and their linking to the existing emergency management centre will also be assessed.

All the urgent population protection measures during a radiation accident at an installation or workplace can be accomplished within the emergency planning zone, as documented by the external emergency plan for the current plant (ETE1,2). [L. 20]. The plan to site ETE3,4 at the Temelín site does not violate the requirements in Article 4b) of Decree No. 215/1997 Coll. [L. 1]. As regards the expected radiation protection level in emergency exposure situations, it can be concluded that population exposure will not exceed the reference levels in the 20 to 100 mSv range, following from International Commission on Radiological Protection recommendations in Publication 103.

2.8.8 CONCLUDING ASSESSMENT

The following conclusions can be made based on the analyses and assessments of the conditions for emergency planning at the site of the ETE3,4 units in accordance with applicable criteria and requirements.

Since the planned building site lies inside the existing protective zone, which is larger than 800 m and in which no permanent residential houses occur, and the internal part of the emergency planning zone is larger than 3 km, the reach of any radiation impacts was found insignificant with regard to the existing situation in the wider area (see Section 4.1.1).4.1.1

The ETE3,4 site enables all the urgent population protection measures during a radiation accident, following from the criterion in Article 4b) of Decree No. 215/1997 Coll. [L. 1], to be rapidly introduced and completely implemented.

No supplementary requirements augmenting the existing design provisions and project basis follow from the analyses presented in Section 2.8.

2.9 SITE ASSESSMENT BASED ON INSTALLATION FACILITY SITING CRITERIA

This section is devoted to the assessment of the properties of the site with respect to existing criteria for the siting of nuclear installations and very significant ionising radiation sources (Article 4 and Article 5 of Decree No. 215/1997 Coll. [L. 1]).

Tab. 151 Site assessment with respect to the exclusion criteria specified in Decree No. 215/1997 Coll.

Art. 4	Exclusion criteria
a)	Expected exceeding of specified mean annual effective radiation doses to individuals in the critical population group present at the site corresponding to the planned siting during operation of the nuclear installation.
	Finding Observed radiation situation at the site, accommodating the 2 running NPP units with VVER1000 reactors (see Section 2.7.2), in combination with the expected activities of discharges from the planned NPP3,4 units and the preliminary assessment of the impacts of the operation of the newly planned installation on the employees, population, and the environment (see Chapter 4) warrants the expectation that the limits of mean annual effective radiation doses, set by applicable legislation (Decree No. 307/2002 Coll. [L. 4]) will not be exceeded. Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph a).
b)	Unfeasibility of a timely introduction and complete implementation of all urgent provisions to protect the population during a radiation accident of an installation or workplace, especially because of the population distribution and existence of residential formations at the site where the siting is planned.
	Finding An emergency preparedness system is in place at the Temelín site in relation to the operation of the NPP3,4 units. This incorporates a fully functional emergency plan which is periodically verified and meets the requirements of the above criterion. Taking into account the requirements for the future design solution specified in the Terms of Reference (see Section 2.8.2.2) it is reasonable to expect the expansion of the Temelín NPP with 2 new PWR units with an output up to 1700 MW to have no impacts on the requirements regarding the definition of the emergency planning zone; the requirements for population protection will not change either, and the feasibility of a timely introduction and full implementation of all urgent population protection measures during a radiation accident at a facility at the site will be confirmed (see Sections 2.8.2 and 2.8.7.1). This conclusion will not be affected by the necessary emergency plan amendments to account for organisational changes in the NPP operation or by the use of some areas included in the emergency plans for the construction of the new NPP 3, 4 units. ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph b).***
c)	Occurrence of karst formations to an extent endangering the stability of the rock massif in the bedrock and overlying rock of the land or area selected for the siting.

Art. 4	Exclusion criteria
	Finding No karst formation or rocks prone to karst formation were identified at the site (see Section 2.6.7.5.2). This can be proved by the textual and graphical documentation of the core drills which were made down to depths of 50 mm at the S1 and S2 site within the plant site survey. No rocks which are prone to karst formation were detected within any of the drills. ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph c).***
d)	Manifestations of post-volcanic activities, such as emanation of gases or seepage of thermal, mineral or mineralised waters on the land or area of the planned siting and at narrower sites.
	Finding Taking into account geological development of the territory (see Section 2.6.2.2.1), no preconditions for the occurrence of the above phenomena exist. This is documented by the results of analysis of the geologic data of the Temelín site, with no marks of occurrence of volcanic rock of the Tertiary and Quaternary Eras or of the above indicators of post-volcanic activity, including recent thermal water springs. ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph d).***
e)	Occurrence of a maximum calculation earthquake intensity 8 on the MSK-64 scale (Medvedev-Sponheuer-Kárník scale assessing the macroseismic effects of earthquakes) on the land where the siting is planned.
	Finding The macroseismic earthquake intensity at the Temelín site, determined by various authors by using different procedures (macroseismic earthquake intensity occurring with a frequency of 1 x 10,000 years), lay in the range of 5.5 to 6.5 degrees on the MSK-64 scale (see Section 2.6.7.1.2). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph e).***
f)	Occurrence of moving and seismically active fractures with simultaneous deformations of the surface of the area and with the potential for the formation of accompanying fractures, as disclosed by geological survey of the site of the planned siting.
	Finding No fractures satisfying the conditions specified in Section 3.6 of IAEA NS-R-3 [L. 14] or phenomena pointing to a potential for such phenomena have been identified at the NPP3,4 building site or in its vicinity (see Section 8.5 of IAEA SSG-9 [L. 6]). Hence, the building site is not exposed to the hazard of displacement on a fracture with manifestation on the surface. No fracture with the potential of moving activity that might imply the occurrence of accompanying fractures with manifestations on the surface of the area was detected in a near vicinity to the building site either. The seismological data collected from the Temelín region give evidence that known strong earthquakes that might induce effects in the landscape are so far from the site that the occurrence of such effects on the NPP3,4 building site is impossible. Moreover, paleoseismological research never detected any such earthquake having occurred at the site during the past 22,000 years as a minimum. The NPP3,4 building site does not fall in the exclusion criterion in Article 4f) of Decree No. 215/1997 Coll. [L. 1] (see Section 2.6.7.2.4). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph f).***

Art. 4	Exclusion criteria
g)	Occurrence of geodynamic phenomena including slides, block slides, plastic displacement of bedrock and fluidisation of soils which endanger the stability of the rock massif at the site selected for siting.
	Finding The physico-mechanical properties of the rocks and soils at the future building NPP3,4 are such that the occurrence of the phenomena listed in the criterion (block slides, plastic displacement of the bedrock) is impossible and the tendency of slopes to landslide is very limited. (See Section 2.6.7.4.2). Furthermore, no soils whose properties point to a tendency to fluidisation were detected by survey at the future NPP3,4 building site (S1 and S2 sites). (See Section 2.6.7.8.2). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph g).***
h)	Occurrence of current or expected deformations of the surface of the areas selected for siting and their narrower sites as a result of gas, oil, or water extraction or underground mining of minerals, application of technologies of dissolution (leaching) of minerals and their extraction, which may endanger the stability of the rock massif in the bedrock and/or overburden of the construction.
	Finding With reference to information from the Raw Material Information Subsystem and results of study of archived documents it is concluded that no gas, oil, or water extraction or underground mining of minerals by the technology of dissolution (leaching) of minerals and their extraction, which may endanger the stability of the rock massif in the bedrock of the future NPP3,4 building site takes place and never took place in the narrower NPP3,4 site area. Also, no underground gas reservoirs exist in the narrower site area (see Section 2.6.7.5.3). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph h).***
i)	Tectonic activity in the narrower area that will, during the operation of the installation or workplace, demonstrably result in a change in the slope of the current surface of the land selected for the siting to an extent exceeding specified technological requirements.
	Finding It follows from the morphological structure of the NPP area as well as from a model of recent dynamics that the future NPP building site is a part of a unified morphostructural block affected by the same extent and nature of vertical motion. A favourable conclusion is also supported by experience from the operation of NPP1 and NPP2: no technological problems such as inclination of the reactor vessel axis were ever experienced when handling the fuel assemblies. (See Section 2.6.7.3.2). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph i).***
j)	Existence of significant groundwater or mineral water reserves at the narrower site where the construction activities or operation of the installation would result in a permanent devaluating change of the water due to radiation effects.

Art. 4	Exclusion criteria
	Finding Review of existing hydrogeological documents gave evidence that the narrower area does not contain used or significant resources of groundwater, whose quality might be endangered by the operation of the new NPP 3,4 units (see Section 2.6.7.10.2). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph j).***
k)	Bearing capacity of soils at the site selected for siting lower than 0.2 MPa and with foundation soils which are sagging and/or highly swelling and/or containing more than 3% organic matter and whose layer thickness is too large to permit soil removal or replacement.
	Finding Engineering-geological survey of the future building site of NPP3,4 did not reveal soils with the above properties at the above extent. The ground of the major buildings of the new NPP3,4 units will be constituted by rock soils with a modulus of deformation higher than 100 MPa on the majority of the foundation base area (see Section 2.6.7.7.2). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph k).***
l)	Existence of geological conditions in the area selected for siting such as water-bearing non-cohesive soil or soft cohesive soil corresponding to tunnel driving category 3 for underground structures.
	This criterion applies to nuclear installations to be sited underground. NPP3,4 is not a nuclear installation of this type.
m)	In the underground construction area, impossibility to cover the main part of the underground structure with a rock massif thicker than triple the largest width of the underground structure, 30 m as a minimum.
	Finding *This criterion applies to nuclear installations to be sited underground. NPP3,4 is not a nuclear installation of this type.*
n)	Existence of old mines at the narrower sites associated with the hazards of undermining, mine water inrush and demolishing effects of major mine or mountain tremor.
	Finding No old mines, associated with the above hazards, were identified at the narrower site of the new reactor units NPP3,4, nor were any signs of past existence of such mines found in historical documents (see Section 2.6.7.5.4) ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph n).***
o)	Mining/extraction of raw materials at the narrower sites that would have adverse impacts on the construction/operation of the installation/workplace.

Art. 4	Exclusion criteria
	Finding According to information in the Raw Material Information Subsystem, no deposits, mines, survey sites, specially protected areas, areas protected for special intervention into the Earth crust or protected deposit areas exist at the narrower site of the future NPP3,4 units. Two plots of non-ore raw material deposits lie in a close vicinity to the narrower site of the NPP3,4 units. Neither the construction nor the operation of the new NPP3,4 units will be affected by the existing extraction of brick-making raw materials at the Bohunice deposit or by the extraction of building stone at the Slavětice quarry (see Section 2.6.7.5). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph o).***
p)	The site encroaching on flood areas of watercourses which are flooded at Q_{100} or on areas that might be flooded as a consequence of a water reservoir/dam accident.
	Finding The site lies beyond the reach of flood areas of watercourses, including water reservoir/dam accidents. The site cannot be impacted by floods arising from storm rainfall to an extent affecting operational safety (see Section 2.5.7). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph p).***
q)	The site encroaching on protected zones of motorways and railways.
	Finding Neither the Temelín NPP area nor the NPP protective zone encroaches on any protective zone of motorways or railways (see Section 2.1.2.2.9). ***Conclusion: the site complies with the requirements of the criterion in Article 4 paragraph q).***

Tab. 152 Site assessment with respect to the conditional exclusion criteria specified in Decree No. 215/1997 Coll.

Art. 5	Conditional criteria
a)	Other karst phenomena that are not specified in Article 4c) hereof and active geodynamic phenomena at the sites selected for siting.
	Finding Analysis of maps, existing literature and field surveys gave evidence that no limestone-type rocks occur at the site. Any known formations lie beyond the boundaries of the narrower site. (See Section 2.6.7.5.2). ***Conclusion: The site complies with the requirements of the criterion in Article 5 paragraph a).*** **No technical provision is required.**
b)	Unfavourable properties of the foundation ground, surrounding soils and rocks at the sites selected for siting.

Art. 5	Conditional criteria
	Finding Engineering-geological survey at the NPP3,4 future building site did not reveal any soils or rocks whose properties might affect unfavourably the foundation or construction of the nuclear installation. The high strength of the rocks at the NPP3,4 site, falling in extractability Class III, may pose a problem. Nevertheless, the consequences of this unfavourable property of the rock massif can be overcome (see Section 2.6.7.7.4). ***Conclusion: The site complies with the requirements of the criterion in Article 5 paragraph b).*** **The provision is technically feasible.**
c)	Maximum calculation earthquake intensities attaining levels 7 to 8 on the MSK-64 scale.
	Finding The macroseismic earthquake intensity at the Temelín site, determined by various authors by using different procedures (macroseismic earthquake intensity occurring with a frequency of 1 x 10,000 years), lay in the range of 5.5 to 6.5 degrees on the MSK-64 scale (see Section **2.6.7.1.2**). ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph c).*** **No technical provision is required.**
d)	Hydrogeological conditions at the building site such as aggravate the monitoring and prediction of the behaviour of groundwater.
	Finding Two groundwater horizons exist at the building site. The hydrogeological structure of the one groundwater horizon is simple, recognisable and monitorable. Only this groundwater horizon is in contact with the construction of the NPP units and some interaction exists. The deep fissure groundwater horizon (50 – 100 m), with no direct relation to the subsurface horizon, is not affected by the NPP construction. Water motion is very slow in that horizon; the groundwater holdup time may be in the order of hundreds to thousands of years. As follows from the study cited in Section 2.6.7.11.2 above, the behaviour of groundwater at the NPP3,4 building site can be monitored and predicted, Hence, the requirement of the criterion specified in Article 5d) of Decree No. 215/1997 Coll. [L. 1] is met. ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph d).*** **No technical provision is required.**
e)	Aggressive groundwater with potential contact with the building structures at the site.
	Finding The degree of aggressiveness of the groundwater at the future NPP building site is typical of the crystalline rock environment. This will mainly play a role in exposure class XA1 (EN 206-1) with regard to the aggressive CO_2 and sulphate ion contents (rarely also XA2 with regard to the aggressive CO_2). Weak aggressiveness is also caused by the acidity of the water (see Section 2.6.7.12.3). Nevertheless, protection of the foundation structures against the aggressive groundwater effects is technically feasible and presenting a proposal for such as solution will be among the requirements put on the contractor. ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph e).*** **The provision is technically feasible.**
f)	Occurrence of well permeable soil and groundwater level at a depth <2 m below the expected level of rough groundwork at the site.

Art. 5	Conditional criteria
	Finding No layers of soil classifiable as "well-permeable" occur at the future building site. The mean filtration factor is k = $2.8.10^{-7}$ $m.s^{-1}$. Slightly higher values are expected for layers of granite rocks and their eluvia. Such layers, however, are spatially limited and do not form a continuous horizon at the site. The groundwater level at the NPP 3,4 building site and in its close vicinity lies at a maximum of 4 to 4.5 m under the ground. This ground is identical with the rough groundwork level. (See Section 2.6.7.11.4). ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph f).*** **The provision is technically feasible.**
g)	High porous or fissure permeability of the rocks as detected by geotechnical survey of underground structures/mines.
	Finding *This criterion applies to nuclear installations to be sited underground. NPP3,4 is not a nuclear installation of this type.*
h)	Geological conditions corresponding to tunnel driving category 2 for underground structures.
	Finding *This criterion applies to nuclear installations to be sited underground. NPP3,4 is not a nuclear installation of this type.*
i)	Extremely unfavourable conditions for the dispersion of atmospheric discharges, mainly due to the morphology of the narrower sites.
	Finding Based on the morphology of the narrower Temelín site as described in Section 2.1.2.1 and the assessment presented in Section 2.4.7, the Temelín site is a plain on which the plant building site is located at an elevation of 3 to 10 m above the surrounding ground. This siting is predisposed to good air motion. No extremely unfavourable atmospheric conditions with respect to dispersion of potentially leaked radioactivity exist at the site. ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph i).*** **No technical provision is required.**
j)	Occurrence of continuously afforested areas at the sites selected for siting, where a potential forest fire could endanger the installation or workplace or their operations or employees.

Art. 5	Conditional criteria
	Finding The new reactor units will be constructed at the industrial area of the fenced and closed premises of the Temelín NPP with grass and ruderal stand. Appreciable forest areas occur to the northwest of the NPP area (approx. 2.5 km) and to the east and northeast at larger distances. Minor stands in the vicinity of the NPP area fence were analysed. From an analysis of the hazard of damage of to NPP3,4 from a fire it was deduced that the consequences of the interaction of a fire with the plant would be negligible (see Section 2.2.7.1). All VAC systems of the operated areas in the safety-related buildings and structures of the NPP3,4 units will be fitted with technological equipment and procedures to ensure inhabitability of the control rooms even if toxic or asphyxiating gases (nitrogen, ammonia) are present in NPP3, 4 outdoor air (see also information to paragraph k) below). This feature will also be activated in the presence of smoke from fire. ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph j).* The provision is technically feasible.**
k)	Presence of industrial production, energy sources, road, rail, and water transport lines, or storage of dangerous substances at the narrower sites, which in unfavourable circumstances might endanger the installation or workplace or their operations or employees.
	Finding Risk sources associated with industrial production, transport routes and hazardous substance pipelines exist within the narrower site of the Temelín NPP and pose non-negligible risk for NPP3,4. The hazards analysed include penetration of toxic clouds of various substances into the VAC system of safety-related operated rooms and other areas, such as control rooms (see Section 2.2.7.2.1). To address this issue, the VAC systems for the new NPP3,4 units will be appropriately designed and the locations of the plant buildings in the building disposition plan will be optimised. The hazard of gas diffusion from the transnational gas pipeline was analysed in Section 2.2.2.5.3.1. A barrier preventing diffusing gas from reaching the NPP1,2 units has been built. This barrier will be modified to protect the NPP3,4 units as well. The new barrier design will be equally efficient against horizontal diffusion. ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph k). The provision is technically feasible.***
l)	Interference of routes and protection zones of gas, oil and product pipelines and underground stockpiles of raw materials into the plots selected for siting,
	Finding A corridor with 3 pipes of the transnational gas pipeline containing pressurised gas exists at a distance of 150 m to the NW from the NPP3,4 site (see the layout in drawing 3 in the Annex Dwg. 3). Article 68 paragraph 2b) of Act No. 458/2000 Coll. [L. 217] specifies a 4 m protective zone for such gas pipelines. The gas pipelines or their protective zones do not encroach on the sites selected for siting the new NPP3,4 reactor units. The safety risks following from the presence of the transnational gas pipeline have been assessed within a dedicated study and are documented in Section 2.2.2.5 above. Measures to prevent horizontal gas diffusion into the area will be required from the contractor. Risks from other events associated with the transnational gas pipeline can be ignored with regard to their minor significance and frequency. ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph l).* No technical provision is required.**

Art. 5	Conditional criteria
m)	Presence of radio and TV transmitters and their protection zones at the sites selected for siting.
	Finding No radio or TV broadcasting transmitters or their protection zones are present at the sites selected for siting the new NPP3,4 units (see a letter from the Czech Telecommunications Office [L. 45]. ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph m).*** **No technical provision is required.**
n)	Protection zones of airports, especially their take-off and landing zones and buildings with ground-based aerial equipment, encroaching on the narrower sites.
	Finding Based on notification from the Civil Aviation Authority dated 4 May 2011, there is no interference between the NPP3,4 site and any airport protecting zone [L. 200]. ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph n).*** **No technical provision is required.**
q)	Possibility of airplane crash with an impact exceeding the strength of the building containing the installation or workplace, with a probability exceeding 10^{-7} yr^{-1}.
	Finding Based on analysis summarised in Section 2.2.2.4 above, the probability of crashing of airplanes in the sports, civil, and military categories by interaction with the new reactor units can exceed 10^{-7}/yr. The requirements that the buildings must withstand a drop of a reference airplane in this category was included in the design requirements so as to comply with the criteria specified in Article 5 of Decree No. 215/1997 Coll. [L. 1] and with the requirements specified in IAEA NS-R-3 [L. 6]. The nuclear safety-related systems, buildings and installations will be designed and built so that the impact of a reference airplane should not disturb their operability. ***Conclusion: The site complies with the requirements of the criterion in Article 4 paragraph q).*** **The provision is technically feasible.**

2.10 DESIGN REQUIREMENTS FOLLOWING FROM THE SITE ASSESSMENT

2.10.1 SCOPE OF THIS SECTION

This section presents a summary of the design requirements that follow from the site assessment in the light of the nuclear installation siting criteria specified in Decree No. 215/1997 Coll. [L. 1] and requirements specified in the IAEA NS-R-3 standard [L. 6]. The terms of reference for the design documentation and construction include some additional design requirements that, although not following immediately from the above criteria/requirements, are related to siting criteria and IAEA standards (such as connection to the grid).

The parts of Section 2.10 reproduced below include those design requirements only that, based on the properties of the site, go beyond the requirements of standards applicable to industrial buildings and structures.

2.10.2 GEOLOGICAL CONDITIONS AND SEISMICITY

2.10.2.1 CONDITIONS FOR THE CONSTRUCTION OF FOUNDATIONS OF THE MAJOR BUILDINGS OF THE TEMELÍN 3 AND 4 REACTOR UNITS

The foundations of the major buildings of the ETE3,4 units will be laid on a non-uniformly weathered bedrock consisting mainly of sillimanitic-biotitic paragneiss and migmatitised biotitic paragneiss with inserts of granitoids (aplite, pegmatite, vein granites) and quartz veins. Detailed information about the occurrence of the various rock types is presented in Section 2.7.2.3.2.

As regards weathering, the rocks at the expected foundation levels will be highly weathered (weathering category 3) or better, i.e. less weathered. The expected deformation modulus on the prevailing area of the foundation bottom is 100 MPa or higher. Sillimanitic-biotitic paragneiss of weathering category 4 (completely weathered) with a modulus of deformation <100 MPa can appear in isolated areas of the foundation bottom, particularly near rigid granitoids or zones of rock disturbance by discontinuities. Such zones will be removed from the foundation bottom and replaced with plain concrete sealing. The extent of such zones will be refined during the next stages of the geological engineering survey. Nevertheless, it can be stated that the occurrence of such rather weathered zones does not preclude the building activity, in other words, it does not fall in the exclusion criterion in Article 5b) of Decree No. 215/1997 [L. 1].

No other unfavourable properties of the foundation soil at the major building foundation level, such as collapsible soils, soil containing organic matter, swelling soils, or soils susceptible to liquefaction, have been detected.

The foundation soil stability is not endangered by the risk of collapse or deformation of the ground surface due to natural conditions (karst cavities) or to human activity (undermining, mineral raw material mining). The building site is not endangered by consequences of slope movements either: neither the morphology of the ground nor the structure of the overburden formations creates preconditions for such phenomena.

At some sites, the construction of foundations for the major buildings of the ETE3,4 units may be aggravated by the following factors (other than hydrogeological):

- High rock strength in some parts of the building site, particularly the S1 area. This unfavourable property of the rock mass can be overcome. Routine approaches include the use of disintegrating hammers, pneumatic picks, or the use of the controlled excavation technology. Procedures inducing a limited seismic load of the surroundings can also be used, such as alternative non-explosive technology. The deployment of such technologies requires coincidence of the construction of ETE3,4 with the operation of ETE1,2.
- At some sites, the buried structures from the ETE3,4 VVER-1000 units under construction may reach the foundation bottom level of the major buildings. Rocks in their surroundings may be affected by secondary weathering. Steps to eliminate such unfavourable properties of the foundation soil will be identical with those applied to rocks affected by natural weathering or damage, i.e. sealing.

2.10.2.2 IMPACTS OF THE BUILDING SITE'S HYDROGEOLOGICAL PROPERTIES ON THE CONDITIONS FOR THE BUILDING OF FOUNDATIONS FOR THE MAJOR BUILDINGS OF THE ETE3,4 UNITS

At the ETE3,4 building site a shallow aquifer bound to Quaternary sediments and the near-the-ground zone of eluvia, largely at the interface of Quaternary sediments with eluvium, or on the base of eluvia and in the subsurface fissure disintegration zone is forming. At some points of the area which is disturbed by the preparation of the NNP building site and NNP1,2 buildings a shallow-lying discontinuous groundwater horizon is forming, being associated with the rather permeable levels of backfills, embankments of various underground building constructions and lines and backfills of building pits. The groundwater level at the NNP3,4 building site and in its close vicinity lies at a maximum of 4 to 4.5 m under the ground (or under the level of rough ground shape).

The survey also provided evidence that the foundation of the major building of the ETE3,4 units will not be affected unfavourably by the occurrence of soils which can be classed as "well" permeable in the sense of the criterion in Article 5g) of Decree No. 215/1997 Coll. [L. 1].

The construction of the ETE3,4 buildings may be aggravated by the precipitation-outflow situation, with many interventions into the natural groundwater outflow. Any difficulties associated with inundation of the construction pits, groundwater level increase or saturation of the backfill around the buildings with water can be avoided by draining the ETE3,4 building site. A specific proposal for the water-draining system shall be included in the next project implementation stages. Moreover, all bottom segments of the buildings will be fitted with an adequate insulation system which is supposed to protect the structures that lie below the ground level against mechanical and chemical action of the (aggressive) groundwater.

The degree of aggressiveness of the groundwater at the future building site is typical of the crystalline rock environment. This will mainly play a role in exposure class XA1 (EN 206-1) with regard to the aggressive CO_2 and sulphate ion contents (rarely also XA2 with regard to the aggressive CO_2). Weak aggressiveness is also caused by the acidity of the water.

2.10.2.3 SEISMIC LOAD OF THE NPP SITE

The seismic load of the NPP site is low: in terms of ground acceleration, SL-2 = 36 cm/s^2 (for the mean value - probability 0.5 and frequency 10E-4). For the mean value and frequency 10E-5, SL-2 = 48 cm/s^2 (see Section 2.6.7.1.6).

The building site is Type 1 as defined in Article 3.1 of the IAEA NS-G-3.6 guide. For details see Section 2.6.2.3.7. The shear (S) wave velocity at the building site exceeds (with a safety margin) the limiting value for Type 1 building sites set by the above IAEA guide, i.e. 1100 $m.s^{-1}$. Hence, dynamic analysis of the response need not be performed.

Additional seismological parameters such as SL-1 (in addition to SL-2) were determined and the response spectrum was generated and the duration of the maximum phase of ground motion was calculated within the specialised study [L. 132].

2.10.3 CLIMATIC CONDITIONS

2.10.3.1 TEMPERATURE

Nuclear safety-related systems, structures and components will be designed for temperatures from the lowest to the highest temperatures expected till 2080 (Tab. 153).

Tab. 153 Air temperature data to be taken into account in the design considerations

Temperature parameter (^{o}C)	Assumption till 2080 (°C)
Highest instantaneous temperature	up to 48
Highest 6-hour average	up to 44
Highest 24-hour average	up to 37
Highest 7-day average	up to 34
Lowest instantaneous temperature	-40
Lowest 6-hour average	-38
Lowest 24-hour average	-30
Lowest 7-day average	-25

2.10.3.2 WIND

The nuclear safety-related systems, structures and components will be designed to withstand wind with parameters specified in Tab. 154 (for a period of 10,000 years).

Tab. 154 Wind parameters to be taken into account in the design considerations

Wind parameter (for a period of 10,000 years)	Speed (m/s)
Wind gust 1 s	65
10-second mean speed	50
10-minute mean speed	38

2.10.3.3 TORNADO

The nuclear safety-related systems, structures and components will be designed to withstand a tornado with parameters in Tab. 155 ("maximum" row).

Tab. 155 Design parameters of a tornado

<table>
<tr><th>Intensity</th><th>Translation velocity estimate</th><th>Maximum wind speed (m/s)</th><th>Translation velocity (m/s)</th><th>Maximum rotary wind velocity (m/s)</th><th>Maximum rotary velocity radius /m)</th><th>Air pressure drop (hPa)</th><th>Air pressure drop rate (hPa/s)</th></tr>
<tr><td rowspan="2">F2</td><td>Mean</td><td>36</td><td>7</td><td>29</td><td rowspan="2">50</td><td rowspan="2">40</td><td>2</td></tr>
<tr><td>Maximum</td><td>67</td><td>13.1</td><td>54</td><td>10</td></tr>
</table>

2.10.3.4 RAIN

The nuclear safety-related systems, structures and components will be designed to withstand rain meeting the parameters specified in Tab. 156.

Tab. 156 Storm rain parameters to be taken into account in the design considerations

Time period	Precipitation level (mm)
15 minutes	42.0
3 hours	66.5
6 hours	83.1
24 hours	109.2

2.10.3.5 SNOW

The nuclear safety-related systems, structures and components will be designed to withstand snow corresponding to the level of a 180 mm water column.

Potential blockage of the VAC suction openings will be avoided by the design of the VAC equipment of the nuclear safety-related systems.

2.10.4 EXTERNAL EFFECTS CAUSED BY HUMANS

2.10.4.1 HORIZONTALLY DIFFUSING LEAKS

Diffusion of gas leaking from the transnational gas pipeline through soil in the horizontal direction under the impermeable ground, whereby the gas could accumulate in the NPP building, explode, and induce fire, will be prevented by installing a barrier in the diffusion pathway.

The buildings of ETE1,2 are currently protected by a diffusion barrier which is referred to as Building Structure 399/01 [L. 18]. This structure will be changed along with the route of the fencing around the guarded area.

2.10.4.2 PROTECTION OF THE ATTENDED AREAS FROM THE PENETRATION OF TOXIC GASES

Attended nuclear safety-related areas, specifically control rooms, will be fitted with technical equipment and procedures to ensure inhabitability if toxic substances (nitric acid vapours, sulphuric acid vapours, hydrazine hydrate vapours, ammonia, or smoke from fire) are present in the NPP3, 4 plant area. For detailed information see Sections 2.2.7.2.1 and 2.3.7.

Provisions at VAC systems will be refined for the specific design solution based on the layout and plans of the plant buildings and air suction points.

2.10.4.3 RESISTANCE OF THE SYSTEMS, STRUCTURES AND COMPONENTS TO THE LOAD FROM A DROPPING AIRPLANE

The nuclear safety-related systems, structures and components will be designed and constructed so that nuclear safety should not be endangered by the drop of a reference airplane (military aircraft 7 tonnes weight, dropping velocity at the moment of contact 200 m/s) in combination with the additional load from the operation.

Provisions at the systems, structures and components will be specified for the specific ETE3,4 design solution (see Section 2.2.7.6.1).

Some of the plant structures may, of course, be designed to withstand crashing of an airplane larger than the reference airplane referred to above.

2.10.4.4 RESTRICTIONS IN THE TRANSNATIONAL GAS PIPELINE SAFETY ZONE

Based on the authority conferred on the transnational gas pipeline owner by Act No. 458/2000 Coll. [L. 217] as regards control over activities within the safety zone and use of the zone's areas, the owner restricted the use of the safety zone area with respect to safety in case of a pipeline accident. The restrictions on the use of the area and requirements for the design solution of the NPP buildings and structures are specified in notification [L. 216]. The safety zone use restrictions are formulated as follows:

- Protective zone, up to 4 m from the pipeline contours: no work is permitted without the written consent of the pipeline operator
- Zone Zero, 4 m to 50 m from the pipeline contours: no building work is permitted without the written consent of the pipeline operator

- Zone A, 51 m to 90 m from the pipeline: specific requirements apply to the siting of buildings and structures, activities, and presence of persons
- Zone B, 91 m to 120 m from the pipeline: it is permitted to store material and site production workplaces of infrequent nature
- Zone C, 121 m to 160 m from the pipeline: siting of buildings and structures with long-term presence of an appreciable number of persons is permitted, with the exception of residential buildings
- Zone D, 160 m to 200 m from the pipeline: no restrictions are put on its use

2.10.5 EMC AND CONNECTION OF THE NPP TO THE GRID

2.10.5.1 ELECTROMAGNETIC COMPATIBILITY

The systems and equipment must perform as designed in an electromagnetic environment meeting the requirements of applicable standards and electromagnetic compatibility (EMC) principles. This concerns, in particular, transient and continuous interference. For details see Section 2.2.2.6.

In particular, the design must define the electromagnetic environment in which the systems/equipment will work. The systems must be designed so as to be able to work in the environment without deterioration of operability due to EMI. Furthermore, the systems/equipment must meet the limits for interference emitted into the electromagnetic environment. Additional provisions as may be appropriate must be in place to address issues of the interface (coupling pathways) between the systems and electromagnetic environment (protections, interference limiters, shielding, equipotentialisation, lightning protection systems, cosmic interference, uninterrupted power supply systems, ...).

2.10.5.2 CONNECTION TO THE GRID

As described in the grid studies referred to in Section 2.2.2.6, the connection of ETE1,2 to the 400 kV and 110 kV network will be modified and, furthermore, the networks themselves will be modified and reinforced so that, for the NPP units, they should:

- ensure evacuation and distribution of the generated power in all states required by the Distribution System Codex
- be stable during disturbances and failures of the grid and at the NPP units (short-term, medium term, long-term dynamics)
- possess an adequate short-circuit resistance, ability to limit and dampen propagation of failures and rapidly and selectively switch off severe failures
- possess an adequate supplying capacity, reliability and short-circuit strength for operational and standby supply to the units

2.10.6 WATER RESERVE WITHIN THE ETE3,4 AREA

In the event of interruption of the service water supply from ČSH at quantities needed to make up all the units' operating circuits, the ETE3,4 units will be shut down immediately. The water reserve at the site and the conditions of its storage will be adequate to cool the reactors in any unit operation mode, including the limiting values for the meteorological parameters, for a period of 30 days or longer, if the conservative analysis of the design solution does not allow a shorter time to be substantiated.

3 CHARACTERISTICS AND PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT IN TERMS OF COMPLIANCE WITH THE NUCLEAR SAFETY REQUIREMENTS WITH IMPACT ON SYSTEMS, STRUCTURES AND COMPONENTS, EQUIPMENT AND SYSTEMS

Chapter 3 of the Initial Safety Analysis Report (ISAR) has been prepared in compliance with the provision of Act No. 18/1997 Coll. [L. 2], Annex A, Part I (2), which stipulates for the permit applicant to prepare the characteristics and preliminary assessment of the design concept in terms of the requirements for the contents of the ISAR as set by the executive legal regulation for nuclear safety, radiation protection and emergency preparedness. The ISAR outline has been conceived in view of the structure pursuant to the guidelines US NRC RG 1.206 "Combined License Applications for Nuclear Power Plants" [L. 275], pursuant to RG 1.206.

Chapter 3 elaborates the principles defined within Section "1.5 Initial Requirements Applied to the Design in Terms of Nuclear Safety", specifically the primary safety goal (see 1.5.1) and basic safety requirements (see 1.5.2).

Within the preliminary consulting with the regulatory authority, the State Office for Nuclear Safety, the following documents were defined as a defined set of binding legislative inputs for the preparation of this Chapter:

- Decree No. 195/1999 Coll., on the requirements for nuclear installations relating to nuclear safety, radiation protection and emergency preparedness [L. 266], hereinafter as the "Decree No. 195/1999 Coll."
- IAEA SSR 2/1, Safety of Nuclear Power Plants: Design, 2012 [L. 252], hereinafter as the "IAEA SSR 2/1"
- WENRA Reactor Safety Reference Levels 2008 [L. 27], hereinafter as the "WENRA" and WENRA Statement on Safety Objectives for New Nuclear Power Plants (November 2010) [L. 270], hereinafter as the "WENRA NEW".
- SÚJB Safety Guide BN-JB-1.0, November 2011 [L. 276]

Chapter 3 is based on the analytical part [L. 279] which used the above-specified legislative inputs to articulate the framework legislative and technical requirements for the design of systems, structures and components and the methods to ensure the stipulated safety and technological functions of individual nuclear installations offered within the tender for the completion of ETE1,2 and ETE3,4.

These requirements were specified using the specific references of the above-specified legislative sources; some of the requirements were articulated in the wider context, possibly based on several legislative sources. As for the references specified, the text does not often include the exact quotation of the source legislative requirements, but their interpretation and application for the purposes of the preparation of ISAR Chapter 3. The legislative and technical requirements are included in the structure of the text within the respective part of Section 3.3 (see the separate introduction) and Sections 3.(4 – 19).1.

Chapter 3 also contains the characteristics of the design for the purposes of preliminary assessment of the design concept which is stipulated by Act No. 18/1997 Coll. [L. 2]. The information for the articulation of the characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant design, however, without the prediction of specific technical designs of individual technological and safety systems.

Respective sections include the partial assessment of whether the preliminary concept of the design in the affected area of the design complies with the specified legislative and technical requirements. The evaluation was carried out in view of general requirements for the functions of systems; the specific method of technological implementation of specific technological units will only be specified in the nuclear power plant design and evaluated in detail at the next stage of the licence documentation. If applicable, the sections also include the articulation of the requirement to respect the design basis based on the conditions of the site, which are specified in Chapter 2 hereof and the most important of them are summarized in Section 2.10. The design characteristics for the purposes of the preliminary assessment are included in the structure of the text within the respective part of Section 3.3 (see the separate introduction) and Sections 3.(4 – 19).2.

The respective part of Section 3.3 (see the separate introduction) and Sections 3.(4 – 19).3 include the formal overview of partial preliminary assessments of the design concept concerning the respective technological system, possibly the design area.

The degree of details of elaboration of individual parts (sections) within Chapter 3 is based on the scope of relevant requirements for the respective design area which are included in the agreed legislative inputs as well as the technical information available in the currently running phase of the tender for the contractor of completion of ETE3,4 at the time of the ISAR preparation. At the next stages of the licence documentation the text shall be further expanded, completed and adjusted.

In case of the preparation of Section “3.15 Analyses of Transient Events and Design Basis Accidents” and “3.19 Probabilistic Analyses and Assessment of Design Extension Conditions” the requirements for the design of safety analyses in accordance with the document IAEA GSR-4, Safety Assessment for Facilities and Activities [L. 278], hereinafter as the “IAEA GSR-4”, were applied in addition to the above-specified legislative inputs, mainly for the following reasons:

- in the latest version of the IAEA standards the requirements for safety analyses were moved to a separate document, i.e. to GSR-4 [L. 278]
- the safety analyses described in the document GSR-4 [L. 278] are immediately bound to the safety level of the ETE3,4 design
- Sections 3.15 and 3.19 are the key parts of the safety report already on the level of the design, and GSR-4 [L. 278] is the relevant safety standard for them

Within the preparation of the whole Chapter 3 the principle of one-time and hierarchical articulation of requirements is consistently applied, which is generally based on the principles defined within Section “1.5.3 Hierarchical Structure of Safety Requirements”. That means that the fundamental general requirements, defined in the introduction of Section “3.3 Design of Structures, Components, Systems and Equipment”, especially in Section “3.3.1 Compliance with Basic Requirements of State Supervision in Order to Provide Safety” are no longer repeatedly explicitly specified in the related sections of Chapter 3, which specify the specific requirements for the systems, structures and components and the performance of technical and safety functions.

The introduction to Section “3.3 Design of Structures, Components, Systems and Equipment” provides further information regarding the structuring of individual sections and the form of their elaboration.

3.1 RESERVE (MAINTAIN NUMBERING IN ACCORDANCE WITH RG 1.206)

3.2 RESERVE (MAINTAIN NUMBERING IN ACCORDANCE WITH RG 1.206)

3.3 DESIGN OF STRUCTURES, COMPONENTS, SYSTEMS AND EQUIPMENT

This integrated part of the Initial Safety Analysis Report is divided into fourteen sections:

The introductory Section “3.3.1 Compliance with Basic Requirements of State Supervision in Order to Provide Safety” is ranked at the top of the hierarchy of Chapter 3 since it includes the definition and articulation of the fundamental and generally applicable design requirements and principles which are applied and elaborated in detail for individual systems, structures and components, safety and technological functions.

Section 3.3.1 is divided into three sub-parts:

The introductory part 3.3.1.1 includes an overview, analysis and specifications of the basic general safety requirements resulting from the arranged sphere of legislation, SÚJB safety guides, IAEA safety requirements and other relevant documents which are to be applied in the design of structures, systems and components important in terms of the safety of the new nuclear installation in the Temelín location.

This sub-part specifies the general principles of the approach to secure safety, defines the barriers to release of radioactive substances and safety functions, characterizes the adopted concept of defence in depth and specifies the approach to its implementation. It also articulates the approach to the design basis and states of the power plant and postulated initiating events, approach to radiation protection and specifies the radiological safety objectives, approach to deterministic and probabilistic acceptance criteria and safety goals of analyses. It further specifies the principles of safety assessment, requirements for the use of proven engineering designs and the reliability of the items important to safety, requirements for the separation and independence of safety systems and applies the criteria for a single failure and elimination of common-cause failures, requirements for the qualification of the equipment and ageing management and for monitoring, testing, maintenance and inspections. The section also includes the characteristics of the approach to the design basis for unit external links and means of communication, characteristics of the approach to apply the human factor in the design, requirements for the physical protection of the power plant and specification of escape routes and specification of the measures for commissioning and decommissioning of the power plant.

The following Section 3.3.1.2 contains the description and specification of basic requirements for systems, structures, components and functions specified in Section 3.3.1.1 These requirements form the envelope of parameters and design requirements in respective areas and they were the basis for the determining of design characteristics for the purposes of preliminary assessment.

The final part, Section 3.3.1.3, contains a comprehensive preliminary assessment of the design concept within the requirements of Section 3.3.1.

The following Sections 3.3.2 – 3.3.13 are elaborated in a unified structure:

The initial part 3.3.(2-13).1 specifies the basic legislative requirements for the respective technological system, possibly the design area.

Sections 3.3.(2-13).2 contain the characteristics prepared for the purposes of preliminary assessment of the design concept for the respective technological system, possibly design area. The information for the specification of the design characteristics was based on the technical part of the BISstipulating the requirements for safety and technical design of the future power plant. The section contains the partial assessment of whether the preliminary concept of the design in the affected area of the design complies with legislative requirements. It is aimed to assess whether the general level of requirements for functioning of structures, components, systems and their equipment was fulfilled. A specific method of technical implementation of those system will only be specified in the NPP design. Subsequently, at the next stage of the safety documentation, the proof of safety relevance of the expected design solution will be submitted.

The final Section 3.3.14 includes the summary preliminary assessment of the design concept for the entire area of design of structures, components, systems and equipment.

The design solution selected for the implementation need not include all systems, structures and components specified within Section 3.3, however, if the resulting design utilizes any of the systems, it shall comply with the relevant requirements.

A detailed information regarding the contents of sub-sections within Section 3.3 is included in the introductions to respective sub-sections.

3.3.1 COMPLIANCE WITH BASIC REQUIREMENTS OF THE STATE SUPERVISION FOR PROVISION OF SAFETY

3.3.1.1 BASIC LEGISLATIVE REQUIREMENTS FOR PROVISION OF SAFETY

The section includes an overview of the basic safety requirements applied on the design by the licence applicant, including the specification of the definition framework. This part was completed to the extent exceeding the arranged framework of the safety report content pursuant to RG 1.206 [L. 275] in order to enable the inspection of whether the BISplaces sufficient emphasis on compliance with the most important basic safety rules and principles, and thus creates the assumption for their full application in the design solution of the future nuclear installation. At the next stage of the licence documentation, the section shall include a brief assessment of the degree of compliance with design requirements for the systems, structures and components important to safety stipulated using the general acceptance criteria by the respective legislation and other documents of the State Office for Nuclear Safety.

3.3.1.1.1 Principles of Safety Approach

Reference: SÚJB Safety Guide BN-JB-1.0 (1), (2), IAEA SSR 2/1 Req. 6, 14, 2.8, 2.10, 2.11, 4.5, 4.6, 4.7, 5.3, 4.8, WENRA App. E 1.1

The design shall ensure the protection of the operating staff, inhabitants and environment against harmful effects of ionising radiation in order to avoid the operation of the NPP to result in a significant contribution to health risks above the framework of the other social risks by applying the following basic principles:

- justification of the design – risks of the design solution of the nuclear installation and method of its utilisation shall be justified and outweighed by its benefit
- protection optimisation – the level of protection shall be optimised in order to achieve the highest level of safety that can be reasonably achieved within the operation of the nuclear installation without unreasonable limiting of its utilisation
- limiting the risks for individuals – it shall be ensured that even in case of justification and optimisation according to the above-specified principles, no individual is threatened by the unacceptable risk of health damage as a result of ionising radiation from the nuclear installation during its life cycle
- prevention and mitigation of effects – all practically feasible efforts shall be exerted to prevent and mitigate the effects of radiation incidents

In order to ensure the highest reasonably achievable level of safety, it will be necessary to adopt the following measures in the nuclear power plant design:

- prevent accidents with harmful effects due to the loss of control over the reactor core and other sources of radiation and mitigate the effects of such accidents should they arise
- make sure, with a high degree of credibility, that the radiation effects are lower than those stipulated by the respective safety objectives and as low as reasonably achievable for all accidents considered in the design
- provide, with a high degree of credibility, extremely low probability of the rise of an accident with serious radiological effects and the maximum practically feasible degree of mitigation of radiological effects of such accident

The design shall comply with all safety requirements of the operator, regulatory body stated in the respective legislation as well as all other relevant codes and standards.

The necessary measures to reduce the exposure to the reasonably achievable level in all operational states and to minimize the occurrence of an accident which might result in the loss of control over the radiation source shall be implemented. Since there is a residual possibility of the occurrence of an accident, the measures to mitigate the effects of the accident are required. Such measures include: inherent safety properties; safety systems; procedures for handling accidents on the part of the operator; external emergency measures on the part of respective authorities. The nuclear power plant design shall be based on the principle that the development of the events which might result in exposure to high doses are practically eliminated and the development of the events with significant probability of occurrence must have either zero or only very small potential radiological effects. The basic goal of the design will be to eliminate the necessity of external emergency measures to limit radiological effects or even to avoid the necessity of such measures in view of the expected results at all, even if such measures are required by the respective legislative documents.

The design shall consider the actual human capabilities and their limitations as well as all factors that might affect human reliability.

The design shall consider the relevant available experience obtained in designing, construction and operation of other power plants and the results of relevant research schemes.

The design shall consider the results of deterministic and probabilistic safety analyses and it shall use the iterative process to ensure that the relevant attention is paid to the prevention of accidents and mitigation of their effects.

The design shall use suitable design measures and operating and eliminating procedures to ensure that generation of radioactive waste and radioactive discharges are at the minimum practically feasible level in terms of both activity and volume.

The design of the items important to safety shall specify the necessary capability, reliability and functionality of the equipment for respective operational states, for accident conditions and for the situations arising due to the internal and external hazards in order to comply with the prescribed acceptance criteria throughout the whole power plant lifetime.

The design basis for each item important to safety shall be substantiated and documented. The documentation shall provide the information necessary for safe operation of the power plant by the operator.

The adequate information regarding the design required for the safe operation and maintenance of the power plant and to enable potential future modifications by the operator shall be provided.

The recommended procedures to be incorporated in the administrative and operating regulations of the power plant (e.g. limits and terms) shall be provided.

3.3.1.1.2 Barriers to Release of Radioactive Substances and Safety Functions

Reference: SÚJB Safety Guide BN-JB-1.0 (13), (3), (5), IAEA SSR 2/1 Req. 4, 4.1, 4.2, WENRA App. E 2.1, 3.1

The physical barriers to release of radioactive substances for the considered pressurized water reactor (in addition to the material structure of the nuclear fuel with high chemical stability and retention capability to prevent the release of fission products) include:

- fuel elements cladding
- reactor coolant pressure boundary
- hermetic containment and its systems

The barriers and measures to maintain the integrity of the barriers shall be constructed in order to

- maintain the integrity of all barriers during the normal and abnormal operation
- maintain the integrity of at least one barrier, i.e. containment, in case of accident states including severe accidents

In order to maintain the functionality of barriers against the release of radioactive substances during the normal and abnormal operation and to the required extent also in accident conditions, the fulfilment of the following fundamental safety functions shall be secured:

- Reactivity control in order to enable the safe shut down of the reactor in any situation and maintaining its sub-critical condition
- Heat removal from the nuclear fuel (both inside and outside the core) for a sufficiently long time, required to maintain the functionality of barriers
- Radiation shielding, keeping all materials inside the physical barriers and preventing the uncontrolled release of radioactive substance to the environment, i.e. limiting the releases in order to avoid exceeding the rated limits in all states considered by the design

The design shall systematically identify inner (inherent) characteristics of the equipment, components and systems, which are inevitable and which contribute to the performance of the fundamental safety functions on the first four levels of defence in depth. The means to monitor the power plant situation aimed to control the performance of the fundamental safety functions shall also be provided.

Securing the fundamental safety functions shall be ensured by the implementation of complementary technical and organisational measures at the different levels of defence in depth. Efficiency and reliability of the measures shall be proved combining the deterministic and probabilistic methods of assessment.

3.3.1.1.3 Concept of Defence in Depth

Reference: SÚJB Safety Guide BN-JB-1.0 (10), IAEA SSR 2/1 2.11, 2.12, 2.13, 2.14

The concept of defence in depth for all safety significant activities and for all power plant states (power operation as well as the states with the reactor shut down) is the primary mean for prevention of accidents and mitigation of their consequences. In compliance with this concept, the protective measures are applied on several independent levels for all relevant activities, so a failure is detected and compensated using appropriate counter-measures. The use of defence in depth in the design and operation provides the protection for expected operational incidents or accidents arising as a result of both equipment failures and human faults both inside the power plant and due to external events.

Applying defence in depth in the design utilizes several levels of various types of measures (inherent physical properties, technical and organisational measures), which are intended either to prevent the events with harmful effects on the environment, or provide adequate protection and mitigation of the effects in the cases when prevention fails. The independent provision or efficiency of each of the protection levels in such a way that a failure of one level does not result in the failure of another level of protection is the inevitable element of defence in depth.

Reference: Decree No. 195/1999 Coll., Article 3, SÚJB Safety Guide BN-JB-1.0 (10), IAEA SSR 2/1 2.14, Req. 7, 4.9, WENRA App. E 2.1

The important part of defence in depth includes the use of the specified multiple physical barriers preventing the propagation of ionising radiation and radionuclides in the environment, and using the technical and organisational measures, used to protect and maintain the efficiency of the barriers, as well as to protect employees and other persons, inhabitants and the environment, on several levels of the system.

Applying the principle of defence in depth ensures that even in the case of multiple equipment or operating staff failures at multiple levels of protection, the population or the environment is be endangered.

The number of designed physical barriers of the nuclear installation depends on the potential source term, efficiency of individual barriers, probability of occurrence and severity of unfavourable internal and external events. In the case of the new nuclear installation, all barriers mentioned in the introduction to Section 3.3.1.1.2 shall be available.

3.3.1.1.4 Approach to Implementation of Defence in Depth

Reference: SÚJB Safety Guide BN-JB-1.0 (12)

The measures of defence in depth shall be hierarchically arranged into the following five levels so that in the case of a failure of the low-level measures the higher-level measures are applied at the next step:

- preventing the deviations from normal operation
- handling the situations of abnormal operation and preventing the transition to accident conditions
- interventions and corrective or protective measures resulting in the avoidance of development or handling of accident conditions, putting the nuclear installation into the safe (shut-down) and controlled state and maintaining at least one efficient barrier to protect the population against the effects of radioactive substances
- handling design extension conditions, including severe accidents, focused on containment or reduction of releases of radioactive substances
- measures to protect the nuclear installation employees in case of a radiation incident and measures to protect the population and the environment in case of a radiation accident

Reference: SÚJB Safety Guide BN-JB-1.0 (11), IAEA SSR 2/1 4.10

The design shall take into consideration that the existence of multiple levels of protection proper cannot be a reason for the continuing power plant operation in the absence of one of the levels. All levels of protection shall be permanently available for normal operation and any deviations from this requirement for specific operational modes shall be justified in order to prove the sufficient compliance with the principle of prevention and mitigation of effects. In case of design basis accidents, a sufficient number of physical barriers shall be preserved for their functioning and efficiency to ensure the fulfilment of the general safety goal.

Reference: SÚJB Safety Guide BN-JB-1.0 (4), IAEA SSR 2/1 4.11

In order to provide the adequate implementation of defence in depth, the design:

- shall utilize multiple physical barriers against the release of radioactive substances into the environment
- shall be conservative (i.e. with sufficient safety margins) and the high-quality implementation shall ensure minimizing of the failures and deviations from normal operation and prevention of accidents to the practicably feasible extent, and preventing a situation when a minor change in a power plant parameter results in a major change in the effects of such parametrical change
- shall ensure such control of the power plant performance (using the inherent properties and technical means) for the failures and/or deviations from the normal operation requiring the intervention of safety systems to be minimized to the maximum practically feasible extent by the design and/or practically eliminated
- shall ensure for the failures and deviations from the normal operation above the framework of the capabilities of the instrumentation and control systemto be handled with a high level of reliability by the automatically launched safety systems; the automatic launch of safety systems shall minimize the need for the operator's interventions in the initial phases of failures and deviations from the normal operation
- shall utilize the technological assemblies, structures, equipment and regulations for the maximum practically feasible handling of the course and limiting of the effects of failures and deviations from the normal operation which exceeded the protective capability of safety systems
- shall use multiple means providing the performance of fundamental safety functions in order to ensure the efficiency of barriers as well as mitigation of the effects of a failure or deviation from the normal operation
- shall preferentially use the inherent and multiple (redundant) safety elements utilizing mainly the negative physical performance feedbacks for the reactivity control; in the case that such elements cannot be used, the systems and components that do not require the external power supply or that, in the case of a power supply loss, remain in the state favourable in terms of safety (safe failure) shall be prioritised

Reference: SÚJB Safety Guide BN-JB-1.0 (10.3), IAEA SSR 2/1 4.12

In order to fulfil the concept of defence in depth, the design shall prevent to the maximum practically feasible degree:

- endangering of physical barrier integrity
- failure of one or multiple barriers
- failure of a barrier as a result of a failure of another one (upstream)
- inacceptable effects of human errors during operation as well as maintenance

Reference: IAEA SSR 2/1 4.13, WENRA E 2.2

The design shall ensure to the maximum practically feasible degree that the first and/or at least the second level of defence in depth is able to prevent the escalation of any failures and/or deviations from normal operation, which are expected to occur during the nuclear power plant lifetime, into an accident.

Reference: IAEA SSR 2/1 Req. 7, WENRA NEW O4

The design shall ensure to the maximum practically feasible degree the independence of the measures on all levels of defence in depth, especially using the diversification of measures.

3.3.1.1.5 Design Basis, Power Plant States and Initiating Events

Reference: SÚJB Safety Guide BN-JB-1.0 (21 - 25), IAEA SSR 2/1 Req. 13, 16,19, 34, 5.1, 5.2, 5.5, 5.6, 5.7, 5.10, 5.24, 5.32, WENRA App. E 4.1, 4.2, 5.1, 6.1

The design basis shall be defined and documented for the design of NPP including mainly:

- nuclear installation states, their categories and respective acceptance criteria
- specification of the functions (namely the safety ones) and requirements for properties of the items important to safety, designed and necessary to provide the safe handling of individual categories of states of the nuclear installation and performing the safety goal as required by the legislation and regulatory authorities
- specific requirements and values (acceptance criteria), representing the marginal limits of the design (upon which the specified functions are met), following from the legislation, requirements of regulatory authorities, generally accepted practice and/or derived from analyses based on calculations or experiments and on the designer’s experience
- in justified cases also the methods of analyses of proofs of nuclear safety and radiation protection and other supporting information

The power plant design shall be based on the requirements for handling the defined list of operational states and accident conditions, including severe accidents. This list shall be used to define the limit conditions to which the nuclear safety-related systems, structures and components shall be designed in order to prove that the safety functions are duly performed and the safety goals are reached. Items, structures and systems important to safety shall handle all defined states reliably and with sufficient safety margins.

Power plant states shall be divided into several categories based on the probability of their occurrence. The specific deterministic acceptance criteria or safety goals as well as the methods of proving that those goals are complied with shall be defined for each category.

Deviations from the normal operation are commonly named the initiating events. The representative spectrum of postulated initiating events shall be specified using the deterministic and probabilistic methods or their combination, while applying operational experience and an engineering estimate. The rise of initiating events shall be considered both during the power plant power operation and in the modes with the reactor shut down, considering the failures in the cooling systems of the power plant, the occurrence of failures in the pools of spent fuel, in the systems for processing and storage of radioactive waste, as well as in handling the nuclear fuel.

The postulated initiating events shall include all possible failures of the power plant equipment, human errors as well as potential failures caused by both internal and external risks, arising during the operation at full capacity, reduced capacity and/or in shut-down modes. Exclusion of any postulated initiating event from the comprehensive list shall be duly technically justified.

First of all, the states and initiating events of internal origin, which are induced by failures of the power plant equipment or faults of the staff, shall be defined.

A group of design basis accidents that the power plant has to handle without exceeding the radiological acceptance criteria shall be singled out from all postulated initiating events. These design basis accidents shall be used to specify the design basis and requirements for the safety systems and other items important to safety which are inevitable for handling accident conditions, with the aim of bringing the power plant to a safe shutdown state and mitigating the effects of accidents.

The design shall also consider handling of very improbable selected design extension conditions which may result from multiple failures of the equipment or faults of the staff and which might potentially result in large or early releases of radioactive substances. The specific requirements for design and acceptance criteria shall be specified for those conditions.

The design extension conditions shall be selected based on the probabilistic safety assessment and with respect to the existing international practice. Also in such case when the probability of the core damage is very low, at least one severe accident related to the melting of the core at low pressure shall be considered in the design. Such accident with a fair probability, which results in the highest load of containment both in terms of its mechanical stressing and in terms of the source element of radioactivity, shall be selected. For that reference accident such measures shall be designed for the systems, structures and components in the power plant design to ensure the adherence to the respective safety goals.

The design shall also consider the specific loads and parameters of the environment acting on the equipment, structures and systems as a result of internal and external effects (risks), which origin in the vicinity, above the framework of the failures of the power plant technological equipment or faults of the staff.

The internal effects include the effects of the jet impact, pipe whips, internal projectiles e.g. from the rupture of rotating machine parts, like turbine blade brake-off, internal flooding, internal fires and explosions, falls and impacts of heavy burdens, failure of pressure parts, braces and other structural parts, electromagnetic interference with the power plant equipment, leakage of water, gas, steam or harmful substances.

Two sub-groups in the group of external effects shall be considered with the assessment of the real possibility of the occurrence specifically for the respective site:

- Natural events: earthquakes, windstorms, lightning, external floods, extreme outside temperatures, extreme rain and snowfalls, ice formation, increase of the underground water level, extreme draught, extreme temperatures of the cooling water and freezing, other risks in the supply of cooling water and air etc.
- Human induced events: external fires and explosions (including a major fire of a gas pipeline), aircraft crashes, electromagnetic interference with the equipment outside the power station, exposure to danger of transport and industrial activities both in the vicinity and on the premises of the power plant (flying objects, gas cloud, explosions), risks from neighbouring buildings and facilities including the neighbouring nuclear installations, propagation of toxic, corrosive or flammable gases etc.

The combinations of burdens from internal initiating events as well as external effects shall also be considered in the power plant design, unless such combinations of events are highly improbable.

Potential failures and damage to the power plant equipment as a result of sabotage (diversionary actions) shall be considered separately.

The specific list of initiating events and their analyses shall be included and analysed at another stage of the safety documentation (in the PSAR). In compiling the specific list the generally considered groups and specific events, as specified in the applicable documents of SÚJB, WENRA and IAEA, shall be taken into consideration.

3.3.1.1.6 Radiation Protection and Radiological Safety Goals

Reference: SÚJB Safety Guide BN-JB-1.0 (2), (3), (22), (39), (127), (136), IAEA SSR 2/1 Req. 5, 2.6, 2.7, 4.3, 4.4, Req. 19, 20, 5.25, 5.31, WENRA App. E 7.1, WENRA NEW App. O2, O3, O6

The design shall ensure the compliance with the requirements of radiation protection pursuant to the special decree (Decree No. 307/2002 Coll. [L. 4]) in order to ensure that in all operational states of the nuclear installation and at all planned discharges of radioactive substances all sources of ionising radiation are subject to administrative and technical control and exposures are kept below the exposure limits, possibly authorized limits on as low a level as reasonably achievable.

The quantitative radiological acceptance criteria and safety goals shall be specified for all power plant states (normal and abnormal operation as well as accident conditions including severe accidents), graded in such a way that the higher the frequency of the occurrence of a specific situation, the more stringent the requirements for its safe handling. The radiological acceptance criteria are expressed using the effective whole-body doses and equivalent doses per selected organs. The events with higher frequency of occurrence shall have negligible or very small radiation effects and the events with severe effects shall be practically eliminated. In order to determine the radiological effects, the process and events arising during the time of the stay of the fuel inside the reactor as well as the states when a part of or the whole fuel is stored in the spent fuel pool shall be considered.

Chapter 4 hereof specifies the binding radiological safety goals for all states of the power plant. For normal and abnormal operation and for the design basis accidents these goals ensure the compliance with the binding dosage limits for the staff and population as stipulated by the respective regulations. In the case of the accidents without the core melting, the design measures shall ensure that they have either no or only minimum radiological effects on the power plant surroundings. Acceptable radiological effects shall be proved using the deterministic conservative method for the accidents without fuel melting. For the minimum radiological effects no urgent measures outside the power plant shall be required (iodine prophylaxis, sheltering or evacuation); for certain scenarios, however, the need to limit the consumption of foodstuffs may arise. For the accidents, including the core meltdown, which cannot be practically eliminated, such design solutions shall be used to limit the need for protective measures for the population both in terms of the affected area and in terms of time (without permanent relocation, without the need for emergency evacuation outside the immediate power plant vicinity, sheltering only to a limited extent, without a long-term restriction of foodstuff consumption), while allowing sufficient time for the implementation of the necessary measures. For the long-term point of view, the period during which the safety functions shall be provided is decisive: depending on the scenario, this may be months and/or even years. The radiological safety goals specified in Chapter 4 hereof are determined independently of the used specific reactor type, its performance and the technical solutions used. These goals are the limit values for all considered reactor technologies, which can thus be used in considerations regarding the power plant location.

3.3.1.1.7 Deterministic and Probabilistic Acceptance Criteria and Safety Goals

Reference: SÚJB Safety Guide BN-JB-1.0 (3), (22), IAEA SSR 2/1 Req. 15, 5.4, 5.25, WENRA E 7.1, 7.2, 7.3, 7.4, 7.5 WENRA NEW O1, O2, O3

After selecting a specific reactor type, the derived technical safety goals shall be specified following the defined radiological criteria and safety goals for all power plant states (normal and abnormal operation as well as accident conditions including severe accidents) so that their observance to the required extent provides the compliance with safety functions and maintains the integrity of barriers against the releases of radioactive substances. Such technical safety goals shall focus on not damaging the fuel system and cladding of fuel elements, integrity of the pressure boundary of the primary and secondary coolant circuits and integrity of the containment.

Analogically, similarly to the radiological criteria and goals, the technical safety goals shall also be graded in such a way that the higher the frequency of the occurrence of a specific situation, the more stringent the requirements for its safe handling. Specifically, the acceptable range of the breach of fuel elements and damage of the fuel system shall be defined depending on the frequency of occurrence of a specific postulated initiating event.

The most important safety goals shall be defined at the following stages of the power plant implementation as the acceptance criteria for safety analyses. The use of design solutions shall ensure that the criteria are not exceeded. The power plant safety shall also be assessed using the probabilistic methods. The probabilistic safety assessment is aimed to prove that the power plant design complies with the existing national and international requirements for nuclear power plant safety in a balanced way. The following probabilistic safety goals are defined for that purpose:

- Summary frequency of major damage to the core shall be reduced to a reasonably achievable level considering all types of hazards, failures and combinations of events. In compliance with the generally acceptable values (IAEA NS-G-1.2, INSAG 12, EUR) the frequency of the major damage to the core zone for a single unit shall not be higher than 1x10-5/year with regard to the power operational states as well as the modes with the reactor shut down, while including the internal influences (nuclear installation risks) and a possibility of disconnecting from the external power grid
- Summary frequency of a large or early release of radioactive substances into the vicinity of the power plant shall not be higher than 1x10-6/year; summary frequency of the events that might results either in an early breach of the primary containment or a large release of radioactive substances, shall be significantly lower than the specified value and such events shall be practically eliminated

For the long-term point of view, the period during which the safety functions shall be provided is decisive: depending on the scenario, this may be months and/or even years.

The above-specified probabilistic safety goals are in compliance with the applicable international recommendations, published in the respective documents of the International Atomic Energy Agency (IAEA, NS-G-1.2; INSAG 12) and EUR. The methodology to be used for the probabilistic assessment in the future shall be in compliance with the internationally accepted procedures. The mean values of the results of the probabilistic safety assessment shall be used for the comparison with probabilistic safety goals.

3.3.1.1.8 Safety Assessment Concepts

Reference: SÚJB Safety Guide BN-JB-1.0 (27), (31), WENRA E 11.1

Nuclear safety shall be proved with relevant credibility using a complex safety design assessment, which is at the same time aimed to identify all radiation sources and assess the potential radiation doses, which can be received by the power plant staff and population as well as the potential environmental effects resulting from the power plant operation. Safety assessment is required for the normal and abnormal operation and for accident conditions included in the design basis of the respective nuclear installation in the respective site after all initiating events with an effect on nuclear safety (including the events induced by natural conditions and events on the site), which cannot be practically eliminated. The assessment shall also include the preparation of evidence documentation (namely for the selected, specially designed equipment), verifying the strength and seismic resistance and lifetime of the equipment.

The safety assessment is based on the results of safety analyses, operational experience, results of relevant research and proven engineering procedures.

Reference: SÚJB Safety Guide BN-JB-1.0 (27), IAEA SSR 2/1 4.17, 5.74

The safety assessment shall be performed starting from the initial phase of the design with subsequent iteration between the designing activities and analytical confirmation of the acceptability of adopted solutions with the scope and details of the assessment gradually increasing.

Reference: SÚJB Safety Guide BN-JB-1.0 (27), IAEA SSR 2/1 Req. 10, 2.9 Req. 42

The safety assessment shall be based on the deterministic and probabilistic safety analyses which form the basis for the determining and confirmation of the design basis for items important to safety. Analyses can be used to specify the design capability to handle the postulated initiating events and accident conditions, demonstrate the efficiency of the items important to safety and to determine the initial conditions for emergency planning.

Reference: IAEA SSR 2/1 5.71, 5.72

The safety analyses also prove that the power plant is designed to be able to comply with the stipulated safety goals for all power plant states and that the conditions of defence in depth are fulfilled.

Reference: SÚJB Safety Guide BN-JB-1.0 (27)

The capability of the equipment to handle operational events and accident conditions shall be verified using deterministic methods. Nuclear safety shall also be assessed for the design extension conditions with the aim of introducing the corresponding preventive and mitigating measures and to limit the exposure of people and the environment even in the cases of events with very low frequency of occurrence. Probabilistic methods shall be used to assess the risk related to the design solution as well as the operation of the nuclear installation. The analyses of the probability of arising of severe accidents, expressed by the annual frequency of occurrence of severe core damage (PSA level 1) will be prepared as well as the analyses of the probability of arising of early radiation accident expressed by the annual frequency of occurrence of early or large radioactive releases, which require the introduction of protective measures to reduce the exposure of persons and the environment, which, however, cannot be implemented in time due to the speed of the development of the events (PSA level 2).

Reference: Decree No. 195/1999 Coll., Article 4 (3), IAEA SSR 2/1 5.74

The quality and suitability of analytical assumptions, methods and calculation programs used for the analyses required for nuclear safety, shall be verified for the respective specific use.

Reference: SÚJB Safety Guide BN-JB-1.0 (28), (35), IAEA SSR 2/1 4.18, 5.73

Proper attention shall be paid to uncertainties in the safety analyses. The analyses shall be documented in a way that enables their independent verification.

3.3.1.1.9 Use of Proven Engineering Solutions

Reference: SÚJB Safety Guide BN-JB-1.0 (19), IAEA SSR 2/1 4.14, WENRA E 9.4

In order to provide reliability and efficiency, the design shall use as much as possible the proven components tried and tested by the experience from their operation in the similar conditions.

Reference: SÚJB Safety Guide BN-JB-1.0 (19), IAEA SSR 2/1 4.16

In case of the use of new technical solutions, their functionality and reliability shall be proved in the documented way using some of the following methods or their combination:

- Using the results of relevant research schemes
- By confirmation tests documenting the compliance with the specified successfulness criteria
- By assessing the relevant operational experience

In justified cases the operational monitoring of the required functional capability of the equipment utilizing the new technological solutions shall be provided.

Reference: SÚJB Safety Guide BN-JB-1.0 (20), IAEA SSR 2/1 4.15

The applied technical regulations, norms, requirements, rules, computation programs used for the activities in the designing process shall be unambiguously specified, their suitability for nuclear installations and compliance with the internationally accepted practice shall be provided and verified. If their combination is used, they shall form a consistent whole and their mutual compatibility and applicability shall be proved. Foreign documents and programs must guarantee such a level of justified interest protection that is identical to that in the Czech Republic.

3.3.1.1.10 Reliability of the Items Important to Safety

Reference: Decree No. 195/1999 Coll., Article 4 (1), SÚJB Safety Guide BN-JB-1.0 (3), IAEA SSR 2/1 Req. 23

The reliable functioning during normal and abnormal operation and in accident conditions shall be provided for buildings, systems and equipment important to safety and radiation protection. The level of reliability shall comply with the safety importance of the equipment.

Reference: Decree No. 195/1999 Coll., Article 4 (2), SÚJB Safety Guide BN-JB-1.0 (14),(44), IAEA SSR 2/1 Req. 23, 5.37, 5.42, WENRA E 9.1, 9.2, 9.4

The reliability of systems shall be achieved by suitable measures, including the use of tried and tested components and their qualification, testing, redundancy, diversity, physical and functional separation. The design shall ensure that the equipment (including the supporting systems) is qualified, procured, installed, commissioned, operated and maintained in order to handle all conditions specified in the design basis with sufficient reliability and efficiency. The limit conditions for which the equipment is designed, shall correspond to the conditions included in the design basis that the equipment is designed to handle.

The systems important to nuclear safety shall be designed in order to enable the system to be transferred into the safe state if a failure of its components is detected using continuously automatically performed diagnostics or the conditions disabling due performance of its safety functions arise. Such threatening conditions mean e.g. switching the protective system sub-systems off, loss of their power supply, or occurrence of predetermined unacceptable states of its operating environment (extremely high or low temperatures, fire, extreme pressure, flooding with water or exposure to steam, high radiation etc.).

Reference: SÚJB Safety Guide BN-JB-1.0 (41), IAEA SSR 2/1 Req. 26, 5.38

When selecting the equipment, the possibilities of both its sudden inclusion and dangerous way of failure development shall be taken into consideration. Equipment with known type of a failure and simple repair and/or replacement shall be preferred.

Reference: IAEA SSR 2/1 Req. 27, 5.43

A failure in supporting systems shall not simultaneously affect various redundant parts of safety systems and/or systems performing various safety functions, or threaten their capability to provide performance of safety functions.

3.3.1.1.11 Separation and Independence of Safety Systems

Reference: SÚJB Safety Guide BN-JB-1.0 (10), (42)

The NPP design shall be sufficiently resistant against a failure of the items important to safety as a result of a single failure and against their common-cause failures. In order to achieve that, the principles of physical separation, functional isolation and independence, redundancy, physical and functional diversity shall be applied.

Reference: Decree No. 195/1999 Coll., Article 19(1)

The protective and control systems shall be separated from each other so that a fault of the control systems does not affect the capability of the protective systems to perform the required safety function. The functionally necessary and practical interconnection of protective and control systems shall be limited and designed so that it does not have a significant effect on nuclear safety.

Reference: Decree No. 195/1999 Coll., Article 19(2)

The protective system shall be designed and set up in order to prevent exceeding of design criteria of the nuclear fuel even in case of faulty functioning of the reactivity control system.

Reference: SÚJB Safety Guide BN-JB-1.0 (45)

The design shall eliminate interference among the items important to safety and it shall ensure that a potential failure or fault of the equipment classified in the lower safety class does not propagate on the equipment classified in the higher safety class. A failure of the systems designed for normal operation shall not affect performance of safety functions.

Reference: IAEA SSR 2/1 Req. 21, 5.33

The interference among various safety systems and/or various redundant elements of one system shall be prevented using a proper application of the measures including physical separation, electrical isolation, functional independence and data transfer independence.

Reference: IAEA SSR 2/1 Req. 40, 5.69

The analysis of potential interference among the systems important to safety shall consider the physical interconnection and potential effects of the operation and/or failure in one system on the operating conditions of other important systems so that the changes in the operating conditions do not affect the required reliable functioning of the components of those systems.

Reference: IAEA SSR 2/1 5.70

If two systems working with the fluid under various pressures are interconnected, then either both systems shall be designed for higher pressure, and/or measures shall be taken to prevent exceeding of the design pressure for the system operating with lower pressure.

Reference: WENRA E 9.2, IAEA SSR 2/1 5.42

Reliability, redundancy, diversity and independence of supporting systems, measures for their isolation and for testing their operability shall comply with the safety significance of the supported system.

3.3.1.1.12 Criterion of a Single Failure, a Common-Cause Failure

Reference: Decree No. 195/1999 Coll., Article 18 (1), SÚJB Safety Guide BN-JB-1.0 (33), (37), (42), IAEA SSR 2/1 Req. 24, 25, 5.39, 5.40, WENRA App. E 8.2, 10.7

The design shall be sufficiently resistant against a failure of items important to safety as a result of a single failure and against their common-cause failures.

The safety systems required for handling design events shall be designed with high functional reliability, redundancy, functional and physical diversity and independence of individual channels so that

- a single failure plus all other failures as a result of such failure do not cause the loss of the system functionality
- disconnecting (putting out of operation) of a component or channel does not result in reduction of the number of independent mutually redundant components or channels to one, unless the acceptable reliability of the protective system operation can be otherwise demonstrated in such case

The above-specified requirements of system redundancy do not apply to the design extension conditions.

A sudden system inclusion shall be assumed as one of the potential failures. It is also necessary to assume the failures of passive components, unless it can be proved with high reliability that the arising of such failure is highly improbable and that the postulated initiating event does not affect the functioning of the component.

Potential causes of common-cause failures which can result in the simultaneus failure of a higher number of components shall be thoroughly analysed and their arising minimized using the principles of diversity, independence, functional isolation and physical separation of the equipment.

3.3.1.1.13 Equipment Qualification

Reference: Decree No. 195/1999 Coll., Article 7, SÚJB Safety Guide BN-JB-1.0 (43)

During normal and abnormal operation, during tests and upon the arising of accident conditions, the items important to safety shall ensure that they are not damaged as a result of failures of other equipment located inside the nuclear installation. That is why they have to be able to withstand the changes of the environment related to such failures and be properly located and adequately protected against dynamic and other effects (thrown objects, line vibrations, fluid leakage, overpressure overload), unless it is proved that such events and their consequences are practically eliminated.

Reference: SÚJB Safety Guide BN-JB-1.0 (19), IAEA SSR 2/1 Req. 30, 5.50, WENRA Issue G 4.1, 4.2

The qualification scheme shall be implemented for items important to safety in order to demonstrate that such items are capable of complying with the requirements for their functionality if required and in the prevailing ambient conditions during their lifetime, also considering the conditions of the equipment maintenance and testing.

Increased attention shall be paid to the qualification of new technologies and equipment.

Reference: IAEA SSR 2/1 5.48

The ambient conditions under consideration shall also include all changes of the environmental conditions expected in the design basis.

Reference: SÚJB Safety Guide BN-JB-1.0 (43), IAEA SSR 2/1 5.49

The qualification scheme shall take account of the effects of ageing caused by various factors of the environment (e.g. vibrations, exposure, humidity and/or temperature) throughout the expected equipment lifetime considering its potential degradation due to expected corrosion, erosion, material fatigue, etc. If the equipment is exposed to external natural events and its activity is required in order to perform a safety function during and after such event, the qualification scheme shall simulate, to the maximum practically achievable degree, the conditions affecting the equipment as a result of a natural event, either using a test and/or analysis or by combining both these methods.

3.3.1.1.14 Requirements for Monitoring, Testing, Maintenance and Inspections

Reference: Decree No. 195/1999 Coll., Article 4 (2), SÚJB Safety Guide BN-JB-1.0 (9), (49), IAEA SSR 2/1 Req. 29

The items important to safety shall be designed in order to provide their qualification, installation commissioning, operation, calibration, functional capability and reliability tests, maintenance, repairs, replacement, inspections and state monitoring using the methods complying with the need to provide their functional capability and integrity in all conditions based on the design basis, including the accident conditions. Technical design of this equipment shall include the safety measures compensating the rise of potential undetected damage during the nuclear installation operation and after postulated initiating events.

Reference: Decree No. 195/1999 Coll., Article 18 (2), IAEA SSR 2/1 Req. 29, 5.46, WENRA App. E 10.8

The items important to safety shall enable safe verification of their availability during the power operation so that their failures that might reduce the functionality or degree of redundancy of those systems are reliably detected. Such verification shall be based mainly on the continuously performed diagnostics of the state of components, their periodical testing and monitoring of redundancy signals. In the case of the components with proven high reliability, periodical testing with the reactor shut-down is also acceptable if there is no practically implementable safe way of providing their testing during operation.

Reference: IAEA SSR 2/1 5.47

If the design of the items important to safety does not enable their testing, inspections and/or monitoring to the required scope, then the alternative sufficiently reliable solution including the following approach shall be used:

- use of other proven alternative or indirect methods including the reference sample checks and/or use of verified and validated calculation methods
- use of increased safety margins and/or other suitable measures established in order to provide the compensation of unexpected failures

Reference: IAEA SSR 2/1 5.45, 5.46

The design shall provide for all activities required for calibration, maintenance, testing, repairs and/or replacement, inspections and monitoring to be carried out using a simplified way in compliance with the relevant national and international standards corresponding to the significance of the safety functions performed without a significant reduction of system availability and without excessive exposure of the staff.

Reference: IAEA SSR 2/1 5.45

The activities required for calibration, testing, maintenance and/or replacement of the items important to safety shall be performed in compliance with the quality standards corresponding to the significance of the safety functions of those items.

Reference: SÚJB Safety Guide BN-JB-1.0 (48)

The limits and conditions for the equipment operation shall include the requirements for its inspections and testing. The frequency of inspections and testing shall be sufficient to verify the reliability and at the same time it shall not result in the excessive degradation of the equipment and unreasonable lapse of lifetime.

3.3.1.1.15 Design Requirements for Unit External Links

Reference: Decree No. 195/1999 Coll., Article 12, IAEA SSR 2/1 Req. 33

A nuclear installation with multiple units should avoid the common use of the safety systems except for the cases when the benefit of such solution to the increase of the safety is proved. The case of arising of accident conditions for one or several units should also be considered as well as a possibility of safe shut-down and cooling of all units.

Reference: IAEA SSR 2/1 5.63

The supporting systems may be shared among the units in order to increase the safety, but such solution shall not increase either the probability or scope of the effects of an accident on any of the units.

Reference: IAEA SSR 2/1 Req. 35

The design of the nuclear power plant connected to the central heating system shall prevent the transport of radioactive substances from the power plant to the heating system in any normal operating, abnormal operating or accident conditions.

Reference: IAEA SSR 2/1 Req. 41

The operability of the items important to safety must not be threatened by failures in the power grid, including the expected changes in the grid voltage and frequency.

3.3.1.1.16 Escape Routes

Reference: IAEA SSR 2/1 Req. 36

The nuclear power plant shall contain a sufficient number of clearly and permanently marked safe escape routes with reliable emergency lighting, ventilation and other services essential for safe use of such escape routes.

Reference: IAEA SSR 2/1 5.64, WENRA S 2.5

The escape routes shall comply with the respective national and international requirements for radiation protection, fire safety, industrial safety and physical protection.

Reference: IAEA SSR 2/1 5.65

At least one escape route from operated premises shall be available after an internal event, external event and/or other combinations of the events considered in the design.

3.3.1.1.17 Means of Communication

Reference: IAEA SSR 2/1 Req. 37, WENRA R 4.2, 4.4

The power plant shall be provided with efficient and resistant means of communication to support safe operation in all modes of normal operation and for use after all postulated initiating events and accident conditions. The design of the means shall enable their regular testing.

Reference: IAEA SSR 2/1 5.66

Suitable warning systems and communication means shall be available in order to provide warning and instructions for all persons present in the power plant and the site during the operational states and in accident conditions.

Reference: IAEA SSR 2/1 5.67, SÚJB Safety Guide BN-JB-1.0 (143)

Suitable diversified means of mutual communication required in terms of the safety shall be available at the power plant, in its immediate vicinity and in respective state authorities. These means shall enable mutual communication of the main control room or another supplementary control centre of the power plant, technical support centre, emergency commission, shelters and physical protection control room, radiation monitoring laboratory, state nuclear regulatory authority, relevant local operating and information centre of the integrated rescue system, local municipal authorities in the emergency planning zone and other affected state authorities.

3.3.1.1.18 Human Factor

Reference: SÚJB Safety Guide BN-JB-1.0 (95), IAEA SSR 2/1 Req. 32

The design shall systematically take account of the issues of human factor and human-machine interface in compliance with the ergonomical principles within the entire power plant design. The design shall support successful activity of the staff in terms of the time available, expected work environment and psychological stress for all considered power plant states. The design shall take account of the fact that the reliability of the action of the staff can only be ensured if the staff has enough time to make decisions and to perform the action, if the required information is unambiguous and if the work environment after the event is acceptable.

The following requirements shall be applied in the design:

Reference: IAEA SSR 2/1 5.53

Specification of a minimum number of operating staff required to carry out the operations performed in parallel after transferring the power plant to the safe shutdown state,

Reference: IAEA SSR 2/1 5.54

Applying the relevant experience of the operating staff by their early engagement to the designing process,

Reference: IAEA SSR 2/1 5.55

Applying the requirements to facilitate staff intervention and minimise human error in the configuration process of the entire power plant and its equipment as well as in the development of the operating regulations including the regulations for maintenance and inspections,

Reference: SÚJB Safety Guide BN-JB-1.0 (95), IAEA SSR 2/1 5.56

Provide such human-machine interface in order to provide the staff with comprehensive, but easily processed information in a way corresponding to time demands for the required decisions and actions,

Reference: SÚJB Safety Guide BN-JB-1.0 (95)

Provide visual or sound indications in order to warn the staff of the deviations from the operational states and processes that might affect the nuclear safety,

Reference: IAEA SSR 2/1 5.57 (a), (c), (d)

Provide the operating staff with the information that enables

- prompt assessment of the power plant state in any condition
- making sure that the design automatic interventions have been made
- specifying suitable manual interventions

Reference: IAEA SSR 2/1 5.57 (b)

Provide the operating staff with such information that is sufficient for safe power plant operation within the framework of limiting parameters for individual systems and equipment and which is also sufficient to make sure that the required intervention can be initiated in a safe way,

Reference: IAEA SSR 2/1 5.58

Provide the conditions for successful intervention of the operating staff by respecting the need for a sufficient time margin for the intervention, expected work environment and psychological demands made on the operator,

Reference: IAEA SSR 2/1 5.59

Minimize the need for prompt operating staff intervention and verify that the operator has enough time to adopt the decision and carry out the intervention. Ensure provision of information required to adopt a decision in a simple and unambiguous way,

Reference: WENRA APP. E9.3

Provide the performance of safety system functions automatically or using passive means so that staff intervention is not required for 30 minutes after the arising of an initiating event; any staff intervention required or necessary during 30 minutes after the arising of the initiating event shall be duly justified,

Reference: IAEA SSR 2/1 5.60

Provide a safe working environment both at the main and the supplementary control room as well as the access route to the supplementary control room,

Reference: IAEA SSR 2/1 5.60, 5.61

Provide such durability and equipment of the main control room as well as the supplementary equipment and supporting workplace to enable the operating staff to perform the activities in compliance with the organizational and technical measures including handling severe accidents according to the instructions of the control and supporting workplace, providing work spaces and work conditions for the operating staff in compliance with the ergonomical principles.

Reference: IAEA SSR 2/1 5.62

Perform verification and validation of all aspects related to the human factor including the use of a simulator in the proper time in order to verify that all required staff interventions have been identified and that those interventions can be carried out correctly.

3.3.1.1.19 Ageing Management

Reference: IAEA SSR 2/1 Req. 31, WENRA I 2.1, SÚJB Safety Guide BN-JB-1.0 (51)

The items important to safety shall have a defined lifetime. The design shall include sufficient margins for ageing mechanisms, neutron embrittlement, wear and tear and other time-dependent mechanisms in order to provide the capability of the equipment to perform the required safety function in the design conditions throughout the whole lifetime. The safety margin shall also take account of the uncertainties of the initial state setting and uncertainties of assessment methods.

Reference: WENRA I 1.1

The ageing management scheme shall be established in order to monitor all mechanisms of ageing relevant for the items important to safety, to specify potential effects of ageing and to specify the inevitable measures to provide availability and reliability of the equipment.

Reference: IAEA SSR 2/1 5.51, WENRA I 2.4

The ageing management scheme shall take account of the effects of ageing and wear and tear for all operational states and conditions of the environment considered for the respective component including the tests, maintenance, maintenance shutdowns as well as the power plant states caused by postulated initiating events.

Reference: SÚJB Safety Guide BN-JB-1.0 (50)

In order to verify the capability of the equipment to perform the required safety function at any stage of its lifetime, the qualification scheme shall be developed during the designing process in order to specify the respective tests before commissioning as well as during the operation proper, also with respect to the maintenance, modifications and testing performed.

Reference: IAEA SSR 2/1 5.52, WENRA I 2.2

The measures for monitoring, testing, sampling and inspections for the assessment of the ageing mechanisms expected by the design shall be adopted in order to provide the identification of unexpected behaviour and/or degradation which might occur during the operation.

Reference: WENRA I 3.1, 3.2

Special attention shall be paid to the monitoring of large structures and components in order to provide timely detection of the ageing effects and enable the implementation of preventive and corrective measures. The reactor pressure vessel ageing management shall take account of all the relevant factors including the embrittlement, thermal ageing and wear in order to compare the actual state with the prediction throughout the whole lifetime.

3.3.1.1.20 Measures for Construction and Decommissioning

Reference: SSR 2/1 Req. 11, 4.19

The items important to safety shall be designed in order to enable manufacturing, construction, assembly, installation and erection in accordance with the prescribed procedures, which ensure achieving the design requirements and required safety characteristics.

The measures for construction and operation shall take account of the respective experience obtained in the construction of similar power plants and related equipment. When adopting the experience from other relevant areas, it shall be shown in advance that such experience is appropriate for the specific nuclear utilization.

Reference: IAEA SSR 2/1 Req.12

The items important to safety shall be designed, manufactured and operated so that their properties and the properties of the surrounding equipment enable and facilitate decontamination, disassembly and replacement during maintenance or removal after the nuclear installation decommissioning. The means and procedures for the implementation of those processes shall be provided sufficiently in advance already at the designing stage.

Reference: IAEA SSR 2/1 4.20

In connection with the measures for decommissioning, the design shall mainly take into account:

- choice of materials in order to practically minimize the amount of radioactive waste to the achievable degree
- required means for handling and provision of the access to the equipment
- availability of the equipment and spaces required for handling and storage of the radioactive waste generated during the operation and measures for handling the radioactive waste generated in the future during the power plant decommissioning

3.3.1.1.21 Physical Protection of the Power Plant

Reference: Decree No. 195/1999 Coll., Article 11, SÚJB Safety Guide BN-JB-1.0 (144), IAEA SSR 2/1 Req. 8, 38, 39, 5.68

The power plant shall be designed in order to provide the required way of the physical protection of the nuclear installation and nuclear material. Within the measures of physical protection, special attention shall be paid to the layout of structures and installation of physical barriers to provide the controlled access of persons and material to the power plant premises with the emphasis placed on prevention of any unauthorized entry and any unauthorized handling (including the IT and software).

The power plant shall be designed to provide increased resistance to sabotage actions, using the power plant safety systems as well as additional technical and organizational measures for physical protection. The measures including the physical separation of systems, redundancy, improved resistance, protective cladding, use of supplementary control room etc. shall be used.

Reference: WENRA NEW O5

It shall, however, be ensured that the measures of nuclear safety and physical protection are designed and implemented in a coordinated way without undesirable mutual limitations.

3.3.1.2 BASIC REQUIREMENTS FOR PROVISION OF SAFETY IN THE PRELIMINARY DESIGN CONCEPT

This section contains the design characteristics in terms of fulfilment of the basic requirements for provision of safety for the purposes of the preliminary design concept assessment. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant. This section includes partial assessment to determine whether the preliminary concept of the design segment in question complies with requirements specified in Sections 3.3.1.1.1 to 3.3.1.1.21. It is aimed to assess, on the general level, the basic safety requirements for the new nuclear installation design, whereas the particular method of fulfilment of those requirements shall be specified in the nuclear power plant design and evaluated at the following stage of safety documentation. In the case of any deviation from the specified requirements the contractor shall submit the proof that the design proposed does not result in the weakening of the general safety level of the presented design solution.

3.3.1.2.1 Principles of the Approach to Provision of Safety in the Preliminary Design Concept

The design solution selected for the new nuclear installation in the Temelín location shall ensure that the power plant staff, environment as well as surrounding population are protected to the sufficient extent against harmful effects of radioactive radiation so that the operation of nuclear units does not result in inadmissible increase of the health risks above the level of other social risks.

Within the EIA process, the assessment of environmental effects of the construction pursuant to the Act No.100/2001 Coll. [L. 255], the comparison of benefits and risks of the prepared design solution and the way of its use was carried out and it was proved that the implementation of the new nuclear installation is justified and its benefits prevail over the effects of potential risks. The results of the assessment confirm the fulfilment of the requirement specified in Section 4 Subsection 2 of the Act No.18/1997 Coll. [L. 2]

The primary principle applied in the future nuclear power plant design shall be to provide the reliable and well-balanced design ensuring the high level of safety. The systems, structures and components of the power plant shall have the required characteristics, parameters and material composition, and they shall be combined and arranged in order to fulfil the operational goals of the power plant while at the same time meet the safety requirements. The design shall enable implementing the operational modes that are essential to fulfil the grid requirements.

By complying with the specified design goals in the area of the doses to the professional staff and population the risk limitation shall be ensured in such a way that no individual is threatened by unacceptable risk of health damage due to ionising radiation from the nuclear installation during its entire life cycle.

The basic goal in nuclear safety shall be to protect individuals, society and the environment by creating and maintaining efficient protection against radiological risks. The safety concept of the design shall be based on applying two basic principles, prevention and limiting of the consequences of accidents.

The design solution shall ensure that within all design conditions the radiation effects are lower than the levels stipulated by respective limits and as low as reasonably achievable and the probability of the arising of an accident with serious radiological consequences is extremely low.

The selected design solution shall be in compliance with the binding legislation, laws, codes and standards applicable for new nuclear installations in the Czech Republic and it shall enable obtaining the licence in the country. The ETE3,4 design shall mainly comply with all binding safety requirements applicable in the Czech Republic, and possibly also other requirements of the operator. In the design preparation, further potential development of the safety requirements during the expected lifetime of the power plant shall be taken into consideration.

By introducing appropriate measures, the limiting of exposure to the reasonably achievable level for all assumed design states shall be achieved. The probability of the arising of accident conditions that might result in the loss of control over the source of ionising radiation shall be minimized.

In the case of arising of accident conditions the power plant shall be equipped with the systems for mitigation of its effects, mainly limiting the release of radionuclides through one or several protective barriers. The utilization of inherent characteristics and simplification of systems is perceived as a significant design goal. The respective minimum time margin shall be required for any operator’s intervention. The optimum use of inherent safety characteristics independent of the power plant control systems or the activity of safety systems (e.g. coefficients of the core reactivity) shall be designed. The operating regulations and instructions to handle the design extension conditions shall be developed with the effective use of available power plant equipment and staff. The adequate means to support the activities of the operating staff, possibly also other staff of the organization of emergency response shall be provided in the design conditions and the design extension conditions (e.g. Technical Support Centre).

The design solution is aimed to eliminate the need for significant extraordinary protective measures affecting the population in case of accident conditions and to enable simplifying of the emergency planning.

For the sake of safety the working spaces for the power plant operating staff shall be designed according to the ergonomic principles while systematically applying the respect to human factor and human-machine interface starting from the early design stage throughout the whole design process. The design shall use the solutions required for the success of the operator’s interventions with regard to the time available, expected physical environment and psychological stress. The necessity of early intervention by the operator shall be minimized. The design solution shall be based on the fact that such intervention is only admissible in cases when it is proved that the operator has enough time to decide and to carry out the intervention, that the information based on which the operator makes his decisions is presented in a simple and unambiguous way and that the physical environment in the main control room or the supplementary control room as well as on the access roads after the event is acceptable.

As a general principle applied in the whole design, the proven components with good history in terms of reliability and integrity shall be used. Applying the experience obtained from the designing, construction and operation of other power plants and from the results of relevant research schemes shall be ensured in such a way that the provided design solution is granted a licence in compliance with the safety requirements either as:

- “Standard design” with the design already pre-licensed (i.e. with the design certification obtained) by the regulatory bodies in the country of origin of the reactor technology or in an EU country or as
- “Reference Plant” with the design already licensed (i.e. with the issued building permit) in the country of origin of the reactor technology or in any European Union country

Proof of safety shall combine the use of deterministic and probabilistic approaches. The deterministic approach shall be used for safety analyses within both the design basis conditions and the design extension conditions. The probabilistic approach shall be used to assess the compliance with the general probabilistic safety goals and to confirm the general balance and consistence of the design. In addition, the probabilistic approach shall be used in the selection of the events to be included in the design extension conditions.

The design shall be designed so that the generation of solid, liquid and gaseous wastes during the lifetime of the planned installation is maintained at such a low level that can be reasonably achieved. This approach shall be based on the obtained operational experience. The power plant systems shall be designed in order to enable the operation in such a way that minimizes the generation of radioactive waste.

The functioning of the individual systems within all power plant modes and conditions as well as the expected system characteristics shall be clearly specified. The design shall apply the procedures for the qualification to confirm that throughout its lifetime the equipment is capable of meeting the requirements for the performance of safety functions for the whole range of expected ambient conditions (e.g. vibrations, temperature, pressure, radiation, humidity) existing before or at the moment of their need. The qualification scheme shall take account of the effects of other environmental factors, e.g. ageing, that may affect the equipment adversely during its required lifetime. In the cases when the equipment is exposed to the effects of external natural events and it is required to implement the safety function during such event or after it, the qualification scheme shall include the conditions to which it is exposed during the natural event.

The operator shall obtain the basic information from the contractor regarding the design and structural solutions within the documentation of the design basis as a background for the process of changes of the design performed by the operator during the future power plant operation.

This procedure shall ensure that the documentation required for the defining of the design basis, technical specifications and requirements for the external power plant links, the actual configuration of the power plant equipment, systems and structures, the procedures for the operation and maintenance and all supporting activities, regardless of the field or source, are available including all details and in an easily accessible form throughout the whole power plant lifetime.
The designing process shall be structured in order to provide the close link between the design and the requirements for structural realizability, availability, maintainability, availability to staff and its professional preparation while respecting the related costs.
The future operator or its representatives shall have access to the information from the processes of designing, construction, commissioning and operation of similar projects within the scope limited by the right to the protected information of the previous customer.

The systematic procedures and instructions for the operation and maintenance shall be developed. These procedures shall include technical specifications, rules and principles for the operation, testing and maintenance, taking account of the human factor as an inseparable part of the human-machine interface in order to reduce the possibility of errors of the operator and improve the operating and maintenance activities.

3.3.1.2.2 Barriers to Release of Radioactive Substances and Safety Functions in the Preliminary Design Concept

Another important concept used within the defence in depth applies to the concept of independent physical barriers preventing the release of radionuclides, i.e.

- fuel rod cladding (1st barrier)
- pressure boundary of the reactor coolant system (2nd barrier)
- containment system (3rd barrier)

Proving the adequacy of the set of barriers is the important part of safety analyses which shall be presented within the preparation of the ETE3,4 design.
Special attention shall be paid to the provision of mutual independence of the barriers, namely the prevention of the simulteneus damage of the second and third barrier.
The first barrier shall be designed in compliance with the expected maximum burn-up. The second and third barrier shall be designed based on the expected lifetime of

the power plant, both for the stationary operational states and for the transients present in the normal operation, abnormal operation as well as the accident conditions.

These barriers shall be designed so that:

- the integrity of all barriers is maintained in the operational states
- during a design basis accident, the integrity of at least one barrier (containment) is maintained

In order to provide the defence in depth, the following fundamental safety functions shall be ensured:

- reactivity control
- heat removal from the reactor core and from stored nuclear fuel
- isolation of radioactive material, shielding against ionising radiation, control of planned discharges and reduction of emergency discharges of radioactive material

The design shall includes such range and structure of systems to enable the ETE3,4 to perform all safety functions in the required scope and maintain the functionality of the barriers against the release of radioactive substances during the normal and abnormal operation and to the required extent also during accident conditions.

The functions of individual systems within all design conditions as well as the system characteristics shall be clearly specified. An appropriate and adequate classification and categorization of the structures, systems and components contributing to the performance of safety functions on the first four levels of defence in depth for ETE3,4 shall be designed, described and applied including the related safety requirements like seismic classification and categorization, environmental classification, system of design marking and QA classification. The required scope of power plant monitoring systems shall be ensured in order to provide the control of performance of the fundamental safety functions.
The design shall include the specification of the requirements essential to specify the set of limits and conditions for the safe power plant operation including:

- defining the limits of operational variables and other important parameters
- setup of the systems performing safety functions
- requirements for the power plant maintenance, testing and inspections in order to ensure that the functioning of the structures, systems and components during their lifetime is appropriate

Compliance with the fundamental safety functions shall be ensured by applying interconnected technical and organizational measures at the different levels of defence in depth. Their efficiency and reliability shall be proved combining the deterministic and probabilistic assessment.

The measures shall be adopted in order to achieve and maintain the required system reliability adequate to the importance of the safety function performed. The design measures to achieve the required reliability of the safety function performance including the redundancy, diversity, independence and physical separation shall be applied.

3.3.1.2.3 Concept of Defence in Depth in the Preliminary Design Concept

The design shall be based on the concept of defence in depth including the sufficient degree of prevention of severe accidents completed with other measures and procedures to handle the design extension conditions including severe accidents. It will be aimed to provide the optimum protection against the release of radioactive substances and to provide the technical basis for the provision of the efficiency of the external emergency plan. In compliance with this concept, the protective measures shall be applied on several independent levels for all relevant activities, so that a failure (even a multiple one) is detected and compensated using appropriate counter-measures.

The safety philosophy of the design shall focus mainly on the prevention of accidents which might result in the potential releases by applying the concept of defence in depth with all safety significant activities and for all power plant states (power operation as well as the modes with the reactor shut down). The aim is to reduce both the probability of the events and its potential effects outside the site and to provide protection under the conditions of expected operational states as well as the accident conditions arising as a result of equipment failures, as well as human faults both inside the power plant and due to external events. The detailed analyses and assessment of the ETE3,4 design shall confirm that the protection of barriers is dimensioned to the sufficient quality and with the required degree of independence, with regard to all regulations and operating procedures.

In order to compensate the potential human failures and equipment failures the design shall utilize several levels of the measures of different type (e.g. inherent physical properties, technical and organizational measures), aimed either to prevent the events with negative external effects or to provide adequate protection and mitigation of the effects in the case that the prevention fails.

The design shall be resistant to the arising of failures on all levels of protection in any state of the reactor. The design measures shall be used to ensure that the safety functions are fulfilled and thus the safety goals are achieved and acceptance criteria complied with. The independent provision of efficiency of each of the protection levels in such way that a failure of one level does not result in the failure of another level of protection is the important element of defence in depth.

3.3.1.2.4 Approach to the Implementation of Defence in Depth in the Preliminary Design Concept

The principle of defence in depth shall include the following five levels:

- prevention of deviations from normal operation
- detection of deviations from normal operation and provision of the measures to prevent the transition to accident conditions
- provision of corrective and protective measures to control and mitigate the accident conditions
- prevention and mitigation of the effects of design extension conditions including severe accidents which could not be included in the design basis previously. Realistic assumptions and the best estimation methods shall be used to analyse the conditions
- mitigation of radiological effects of potential releases of radioactive materials

In case of a failure of lower-level measures, the higher-level measure shall be applied in the following step.

The need to ensure that the risks following from the operation of ETE3,4 are acceptably low, shall be reflected at all stages from the design to the final decommissioning of the power plant and it shall be fulfilled with the sum of the many following measures providing the implementation of the defence in depth concept:

- the design shall be designed in order to prevent, to the reasonably achievable extent:
    - endangering of physical barrier integrity
    - failure of a barrier, in case of its endangering
    - breach of a barrier as a result of a failure of another (upstream) barrier
    - unacceptable effects of human errors during operation as well as maintenance
- the design shall consider the fact that the existence of multiple levels of protection is not a sufficient basis for further power plant operation in the case of the absence of one level of protection. All levels of protection shall be available anytime during the whole normal operation, while justified exceptions can be specified for some other operational modes
- a sufficient number of physical barriers shall be maintained for the operational states and design basis accidents so that their functioning and efficiency provide the fulfilment of safety goals
- systematic identification and detailed analyses of all design conditions shall be performed in order to determine the functional requirements for the equipment
- the design shall be designed conservatively, with sufficient safety margins; high standards shall be applied in the design and manufacture of the equipment required to achieve the high safety level (using adequate standards), strengthened by the corresponding level of quality assurance and independent control aimed to ensure that all safety requirements are fulfilled and the situation, when a minor change of a parameter results in an extensive change as a result of such parametrical change, is prevented
- full justification of the power plant design shall be based on the adequate theoretical and experimental basis
- if practicable, the used components shall be verified to the sufficient extent in the previous designs of light water reactors,
- in order to provide safety functions, the separate, diverse and redundant systems and equipment shall be used
- attention shall be paid to the power plant layout in order to minimize the radiation exposure of the operating staff during the normal operation and to reduce propagation of potential radioactive releases in accident conditions
- multiple barriers aimed to limit propagation of the radioactive substances shall be set

- the solutions clearly identifying the types of failures of systems, which perform safety functions, and decreasing their occurrence to the acceptably low level shall be applied systematically
- the adequacy of the measures used in the failure sequences and event trees, in reliability analyses and integrity analyses (including the inclusion of common-cause failures) shall be proved
- resistance to a single failure shall be verified using appropriate studies of failure states supported by the research and development, where relevant
- the design shall preferentially use the elements utilizing the negative physical performance feedback for the reactivity control; if such elements cannot be used, the systems and equipment, which do not require external power supply or which, in case of its loss, remain in the state favourable in terms of safety (safe failure) shall be prioritised
- the equipment shall be designed and constructed in compliance with the design so that it is capable of performing the required purpose
- the systematic scheme of maintenance, supervision and operating inspections specified by the documented procedures shall be provided
- the efficient ageing management scheme shall be adopted to be achieved via coordination of the existing schemes including maintenance, operating inspections, the supervision scheme as well as using the operating and technical support and external schemes including the research and development schemes
- control, metering and signalling equipment shall be provided in order to support the trained operating staff in any conditions
- the equipment to provide the systematic scheme of the environmental impact assessment shall be available throughout the power plant's lifetime
- the systems, structures and components including the regulations and instructions to handle the course and reduce the effects of failures and deviations from the normal operation which exceeded the protective capacity of the safety systems (design extension conditions), shall be applied
- monitoring, records, reporting and analysis of abnormal states and accident conditions of the power plant shall be provided throughout its whole lifetime in order to prevent the failures and confirm that the power plant reliability does not contradict the assumptions adopted in the reliability analyses
- such control of the power plant performance (using the inherent properties and technical means) shall be provided in order to ensure that the failures or deviations from the normal operation requiring the intervention of safety systems are minimized by the design to the maximum extent or practically eliminated

The design shall minimize the power plant sensitivity to the faults or inactivity of an operator or maintenance staff. The requirements for the operator’s intervention shall be minimized, simplified or postponed.

The design measures shall include, if it is reasonably practicable:

- automatic activation of unit protection systems if the power plant parameters exceed the limits of the normal operation or the values of the protection system setup
- measures increasing the probability that the operator will be able to remedy the faults
- measures to prolong the time that the operators have in order to recover the activities of auxiliary systems and power supply

The time between the rise of a deviation from the normal operation and any serious effects following from the absence of operator's interventions should be as long as reasonably achievable.

The design solution shall ensure that the first, and/or the second level of defence in depth, in the last resort, prevents, with maximum probability, the development of failures or deviations from the normal operation into accident conditions.

Diversity shall be required for important systems or functions in order to achieve the stipulated target values of reliability and to fulfil the concept of defence in depth. The diversity requirements shall be fulfilled by combining the active and passive elements. The safety categorization of the equipment providing the diversity shall be specified in the design. The diversity principle shall be applied for the redundant systems or components that perform the same safety functions, by utilizing a different construction of those systems or components. The differences in construction may include different principles of functioning, work with different physical values, use of different operational conditions, production by different manufacturers, etc.

The design shall consider the possibility that the requirement for the action of any level of protection may arise in a random state of the reactor and the design measures shall be designed in order to ensure that the safety functions are fulfilled so that the safety goals and acceptance criteria are met.

3.3.1.2.5 Design Basis, Power Plant States and Initiating Events in the Preliminary Design Concept

Within the design, the design basis shall be specified and documented including the defining of power plant conditions (states), specifying the functions and properties of the items important to safety, acceptance criteria defining the design boundaries within which the safety functions are performed, and methods of proofs of nuclear safety used in the analyses.

The design solution of the power plant shall include the list of operational states and design basis accidents taking account of the design specifics that shall be verified using the power plant PSA.

A full list of postulated initiating events of internal origin shall be developed with the events adequately classified based on the expected frequency of their occurrence or based on the current practice for similar power plants. These categories are as follows:

- normal operation,
- abnormal operation

- design basis accidents with low frequency of occurrence
- design basis accidents with very low frequency of occurrence

The initiating events shall take account of the range of possible reactor states including the failures in case of reactor shut-down states.

In each category the so-called “covering” events, which include all other states/events in the same category in terms of the effects and which shall be assessed based on the acceptance criteria, shall be selected.

The design of structures, systems and components shall take account of the load following from the combinations of the normal and extreme conditions of the outside environment, initiating events and risks unless such combinations are highly improbable. The structures, systems and components shall be designed and dimensioned in order to endure the effects of the selected range of events with sufficient safety margins.

The design shall be designed so that an accident does not result in the loss of systems which are required to mitigate the effects of the accident. The systems of protections shall be designed as independent of the control (regulation) systems used in normal operation.

The criteria to ensure the integrity of safety barriers or efficiency of the safety functions shall be set for the deterministic safety assessment.

In the case of the majority of design basis accidents, the reactor shall be shut down and transferred to the subcritical state. If the reactor criticality and increase of the power might reoccur, e.g. as a result of cooling after the break of the main steam line, the power level shall be reduced in order to avoid breaching the limits for fuel. At the same time it shall be ensured that the clear power plant state indication is available to provide the operator with a possibility to transfer the power plant to the safe shutdown state in accordance with the operating regulations.

During the design process the requirements necessary to determine the set of limits and conditions of the safe power plant operation shall be specified, including:

- setup of the limits of action in case of the changes of important parameters
- setup of the systems performing safety functions
- requirements for maintenance, testing and control of the power plant in order to ensure that the functioning of the structures, systems and components during their lifetime meets the design assumptions

The characteristics of the main nuclear steam supply system (NSSS) including the parameters of the reactor core, compensator capacity, water reserve in the steam generator etc. shall have sufficient margins in order to ensure that the safety and operational goals are met.

Each event of the abnormal operation and design basis accident shall be classified in the respective category depending on the probability of the initiating event and its significance. For each category of design basis conditions a limit exposure dose shall be set and other safety goals specified, including the method of proving that those goals are met.

The design conditions for the equipment shall be chosen from the envelope of initiating events potentially arising both during the nuclear installation power operation and during the states with the reactor shut-down. In the process, the failures in the reactor coolant systems as well as the failures of the systems of the spent fuel pool, failures of the systems for radioactive waste processing or storage and failures during nuclear fuel handling shall be considered. Both the failures of power plant equipment and human faults and failures due to internal as well as external risks shall be considered. The list of postulated initiating events shall be specified using the deterministic and probabilistic methods and their combining, with regard to the operational experience and engineering estimate.

The comprehensive list of postulated initiating events shall include the group of design basis accidents that the nuclear unit has to handle without exceeding the radiological acceptance criteria. The analyses of those design basis accidents shall be used to specify the design basis and requirements for the functionality of safety systems and other items important to safety which are necessary to handle the accident conditions, with the aim of bringing the power plant to a safe shutdown state and mitigating the effects of accidents.

The design of ETE3,4 shall also include the design extension conditions including severe accidents with the specific design requirements and radiological acceptance criteria specified.

Specific equipment and systems (and their design limits) shall be specified in order to handle those conditions.

In connection with the design extension conditions, the design shall provide compliance with the following three goals:

- Reduce the probability or mitigate the effects of complex events like ATWS and SBO, which include multiple failures exceeding those considered in the deterministic design basis
- Prevent containment failure. The aim is to prevent, to the reasonably achievable degree, all severe accidents that might result in containment failure. For the sequences resulting in severe damage of the fuel system which might result in containment failure it shall be proved that their probability is so low that they do not need to be considered in the draft design
- Reduce the effects of severe accidents. The containment shall be capable of handling severe damage of the fuel system, by containing and cooling of the remainders of the fuel system, preventing the interaction of the melt and concrete, reducing the releases, reducing the load related to the fuel cladding oxidizing and hydrogen burning and prolonging the time required for the operator’s intervention needed for accident management.

Fulfilment of those goals shall be proved using deterministic methods.

When setting the range of design extension conditions, the requirement shall also include the compliance with the probabilistic safety goals. The probabilistic assessment (together with technical assessment and applying other criteria) shall also be used to specify the list of design extension conditions and to design the related design measures.
That way the basic list of design extension conditions shall be specified so the design shall consider the severe accidents with the largest contribution to the frequency of releases. The best-estimate methods shall be used in their assessment.

The ETE3,4 design shall be based on the relevant design basis based on the conditions of the site, which are specified in Chapter 2 hereof and the most important of them summarized in Section 2.10, and it shall be executed with the capability to handle the internal and external risks. The design shall consider the specific loads and parameters of the environment affecting the equipment, structures and systems as a result of internal and external risks. The following shall be considered:

- internal effects see 3.3.5
- external natural effects see 3.3.3
- external effects caused by humans see 3.3.3

All potential events relevant to the site including the failures and damage to the power plant equipment as a result of sabotage shall be assessed as external effects.

3.3.1.2.6 Radiation Protection and Radiological Safety Goals in the Preliminary Design Concept

The specific goal of the radiation protection is to ensure that, in all states, the radiation exposure from the power plant and radiation dose from any radioactive material release from the power plant are lower than the respective limits and as low as reasonably achievable (ALARA).
The basic principle to provide the radiation protection of the operating staff applied in the design shall be that the individual doses shall be ALARA, see ICRP 60/91, ICRP 103/2007, IAEA Safety Series 9/82 and IAEA Basic Safety Standards, Euroatom Directive 96/29.

The power plant design shall enable performing the entire required scope of maintenance while keeping the optimised balance between the reduction of individual doses and maintainability requirements.

It shall be ensured that the requirements stipulated by the Czech law in radiation protection (see the Decree No. 307/2002 Coll. [L. 4]) are complied with so that in all operational states of the nuclear installation and during all planned discharges of radioactive substances all sources of ionising radiation are under control and exposures are kept below the exposure limits, possibly authorized limits and on the level that is as low as reasonably achievable.

The quantitative radiological acceptance criteria and safety goals shall be specified for all power plant states, graded in such a way that the higher the frequency of the occurrence of a specific situation, the more stringent the requirements for its safe handling. These criteria and goals shall be specified for normal operation, abnormal operation, design basis accidents and design extension conditions. In specifying the radiological effects the cases of the fuel stored in the spent fuel pool shall also be taken into consideration.

The limit values of occupational doses stipulated in terms of the level of radiation protection shall be in compliance with the values specified in Chapter 4 hereof. The assessment shall be made in order to prove that the above-specified dose targets are met.

Not exceeding of these limit values shall be provided using suitable construction of systems and choice of materials (e.g. choice of materials and chemical composition of the coolant for lower contamination, reduced radiation fields, improved conditions for periodical and random maintenance), layout (shielding and distance) and via the organization of operation and maintenance (e.g. maintenance focused on reliability and use of automation).

The effective doses of an individual from the population resulting from direct exposure and committed effective dose from internal exposure in the conditions of normal and abnormal operation shall not exceed the values specified in Chapter 4.

The designer specifies the radiological criteria for radioactive discharges and releases in the structure specified in Chapter 4 hereof, for both normal and abnormal operation.
The design shall specify the maximum value of releases for design basis accidents and design extension conditions including severe accidents in compliance with the values specified in Chapter 4.

Conformance to these goals shall provide the design compliance with the binding dose limits for the employees and population stipulated in the respective regulations for the conditions of normal and abnormal operation and in the conditions of design basis accidents.

The goals for the design extension conditions are specified so that there is only the the limited extent of the necessary protective measures outside the power plant area proper and there is no need to expand the existing zone of emergency planning (external).

The design solution shall ensure that the design basis conditions without the total melt of the fuel system result in either no or only very low radiological effects in the power plant surroundings. The acceptability of radiological effects in the case of accidents without heavy damage to the fuel system shall be proved deterministically in the conservative way.
When assessing the observance of the radiological goals, the best current practice shall be considered in specifying the degree of releases from fuel, level of activity of reactor coolant, level of releases from the reactor coolant system, etc.

The radiological safety goals specified in Chapter 4 hereof are determined independently of the used specific reactor type, its performance and the technical solutions used. These goals are the limit values for all considered reactor technologies, which can thus be used in considerations regarding the power plant location. Each of the reactor technologies considered shall ensure the observance of the specified exposure limits and shall provably respect the basic principle of radiation protection, i.e. to limit the exposure to the level as low as reasonably achievable (the ALARA principle) during the operational states as well as in accident conditions. The required scope of the technical and organizational measures for the Temelín location was preliminarily (using the covering way) assessed in the study of environmental effects and shall be specified within the PSAR preparation.

3.3.1.2.7 Deterministic and Probabilistic Acceptance Criteria and Safety Goals in the Preliminary Design Concept

The design shall specify the derived technical safety goals the observance of which in all states of the nuclear power plant shall ensure the fulfilment of safety functions and thus the preservation of the integrity of barriers against the releases of radioactive substances. These technical safety goals shall focus on preserving the integrity of the fuel system, integrity of the pressure boundary of the primary and secondary coolant circuits and integrity of the containment. These derived safety goals shall be graded in such a way that the higher the frequency of the occurrence of a specific situation, the more stringent the requirements for its safe handling. The acceptable range of the breach of the fuel system shall be defined depending on the frequency of occurrence of a specific postulated initiating event. The most important of these derived safety goals shall be stipulated as the acceptance criteria of safety analyses.

The maximum acceptable releases from the containment system shall be specified both for the design basis conditions and for the design extension conditions. The safety analyses shall prove that the radiological criteria are met.

The probabilistic safety goals shall be specified for the summary frequency of the severe core damage, summary frequency of exceeding the radiation acceptance criteria for the design extension conditions and the summary frequency of the sequences including large and early releases.

In compliance with the safety approach applied in the design, the following probabilistic quantitative design goals are defined:

- summary frequency of the severe core damage shall be lower than 10^{-5} reactor year^{-1}
- summary frequency of exceeding the radiation acceptance criteria for the design extension conditions shall be lower than 10^{-6} reactor year^{-1}
- summary frequency of early (early failure of containment system) or large radioactive releases (one order higher than the radiation acceptance criteria for design extension conditions) shall be sufficiently below 10^{-6}/year

3.3.1.2.8 Safety Assessment Concepts in the Preliminary Design Concept

The design safety shall be proved using the respective safety assessment. It shall be aimed at identifying the radiation sources and assessing potential radiation doses obtained by the power plant staff and population including their environmental impacts. The safety assessment shall be performed for the conditions of normal and abnormal operation and for accident conditions included in the power plant design basis after all initiating events with the impact on nuclear safety (including the events induced by natural conditions and events on the site), which cannot be practically eliminated. Within the assessment the evidence documentation shall be elaborated (mainly for the selected equipment), verifying both the strength and seismic resistance and lifetime of the equipment.

The safety assessment based on the results of safety analyses shall be presented within the developed PSAR, which shall be based on the previous operational experience, results of research studies and proven engineering procedures.

During the design phase, the safety assessment system shall be established providing the mutual functional inspections and assessment of the design with respect to the probabilistic safety analyses, readiness and reliability analyses enabling to modify the design so that mainly its safety goals are met. The safety assessment shall combine the use of deterministic and probabilistic methods. It shall be proved that the combination of design basis conditions and design extension conditions used in the design enables the nuclear power plant to meet the probabilistic goals and it shall demonstrate the design capability to handle the postulated initiating events and accident conditions, efficiency of the items important for nuclear safety and it shall be possible to specify the input conditions for emergency planning.

The probabilistic assessment shall be used to assess the compliance with the general probabilistic safety goals and it provides the assurance regarding the general balance and consistence of the design. Besides, the probabilistic assessment shall provide the information for the selection of design extension conditions.

The deterministic assessment shall be used for safety analyses of design basis conditions and design extension conditions.

The design shall be designed in compliance with deterministic criteria and verified using deterministic assessment considering the full list of internal conditions and risks as well as the design extension conditions. It shall be proved that the power plant is designed to be able to meet all the stipulated limits for radioactive releases and for potential radiation doses in all considered nuclear installation states and that the conditions of defence in depth are met. The power plant safety shall also be confirmed in the design extension conditions with the aim of applying the appropriate preventive and mitigating measures and to limit the exposure or persons and environment even in case of the events with very low frequency of occurrence.

PSA shall be developed both on Level 1 (determining the frequency of events resulting in the severe core damage) and on Level 2 (determining the frequency of large and early releases).

The PSA study shall consider all design states listed among the design basis conditions in Section 3.15 hereof. For the analytical basis, methods and computation programs used for the analyses important to safety, the quality and appropriateness shall be verified for their specific use. In the analyses, attention shall be paid to the uncertainties. The analyses shall be documented in such a way that enables their subsequent independent verification.

3.3.1.2.9 Use of Proven Engineering Solutions in the Preliminary Design Concept

The power plant solution shall be based on the type design licensable (or already licensed) in the country of origin of the used technology of the nuclear installation or in an EU state.

The design shall use proven technologies. The proven technology is defined as the technology that uses proven materials and technical solutions used successfully in the analogical working conditions (pressure level, chemical composition of the environment, etc.) in the existing pressurized water reactors or in similar operating environments and applications in other industries (for several years at least). Proven components with a good history of reliability and integrity shall be used.

Innovative solutions shall only be chosen in the case that they provide a clear advantage in one or more specific areas (e.g. safety, price, output, reliability) without negative effects that might affect other areas. Their use shall be substantiated by the use of the results of relevant research schemes or the experimental verification assessment.

Nevertheless, it is expected that all innovations that are beneficial in terms of the safety increase shall be taken into consideration. In such cases, if the innovative technologies are used, the operability of the equipment or system shall be verified under the conditions corresponding to the operation or testing in the power plant.

Section 1.5 hereof specifies the requirements for the applied technical regulations, norms and rules for the designing activities together with assurance and verification of their applicability to nuclear installations and compliance with the internationally accepted practice. If the combination of regulations of different origin is used, such combination shall form the consistent unit while proving mutual compatibility of various regulations.

The respective specifications, design drawings and system descriptions are prepared for systems, structures and components documenting compliance with the acceptance criteria of the highest level and other design requirements, computations, production, constructions and testing.

The design shall meet the respective national or international codes and standards and the power plant shall meet the requirements of the best applicable standards for production, construction, controls, maintenance and operation, complying with both the safety classification and applicable requirements for the reliability of the power plant and its components.

3.3.1.2.10 Reliability of the Items Important to Safety in the Preliminary Design Concept

The design shall adopt the measures to achieve and maintain the required reliability of structures, systems and components that are adequate to the importance of the safety function performed in the conditions of normal and abnormal operation and in accident conditions.

The design shall use appropriate measures including the proven and qualified components, their testing, redundancy, diversity, physical and functional separation providing the required degree of reliability. The equipment shall be qualified, provided, installed, commissioned, operated and maintained in order to handle all conditions specified in the design basis with sufficient reliability and efficiency. The limit conditions, for which the systems, structures and components are designed, shall match the conditions included in the design basis that the equipment is designed to handle. The fulfilment of specified safety functions including the appropriate degree of redundancy shall be proved.

In order to increase the reliability and to meet the criterion of a single failure in the systems performing the safety functions, the adequate degree of redundancy shall be used.

Identical or different components may be used to provide redundancy.

The systems important to nuclear safety shall be designed in order to enable the system to be transferred into the safe state if a failure of its components is detected using the continuously automatically performed diagnostics or if the conditions disabling performance of its safety functions rise. Such safety threatening conditions mean e.g. switching the protective system sub-systems off, loss of their power supply, or achieving the predetermined unacceptable levels of its operating environment (extremely high or low temperatures, fire, extreme pressure, flooding with water or exposure to steam, high radiation, etc.).

False activity of the automatics shall be considered one of the possible types of a failure unless the specific measures exist to prevent such activity or such activity can be eliminated as improbable.

In the selection of the equipment, the possibilities of both the sudden inclusion and dangerous type of a failure shall be taken into consideration. The equipment with known type of a failure and simple replacement and repair shall be preferred.

The design shall avoid excessive complexity by utilizing an adequate number of components to achieve the safety function and intercross the redundant branches. The failures in supporting systems must not simultaneously affect various redundant parts of safety systems and/or systems performing safety functions or threaten their capability to provide performance of safety functions.

3.3.1.2.11 Separation and Independence of Safety Systems in the Preliminary Design Concept

In some systems required for the performance of safety functions, the principle of functional independence and diversity shall be applied in order to achieve the stipulated reliability values and to fulfil the concept of defence in depth. The diversity requirements can be achieved by combining the active and passive elements. In suitable cases the diversity principle shall be applied for the redundant systems or components that perform the same safety function, by incorporating different properties in those systems or components. Such properties may include different operating principles, different physical parameters, different operating conditions, production by different manufacturers, etc.

The following principles of functional independence shall be applied in the design:

- independence of functional branches of redundant systems to the reasonably achievable level and in all cases when it means a general benefit to the safety
- independence between components and effects of initiating events; the initiating event should not cause a failure or a loss of the safety function required to mitigate the event
- independence between the components of various safety categories so that a member of a higher safety category could not be endangered by a failure of an element of a lower safety category
- maximum mutual independence of the elements of the highest safety category

In order to achieve the required degree of independence, the design shall apply the functional and/or physical separation.

The functional separation shall be provided in order to reduce the probability of unfavourable interactions between the equipment and components of the redundant or supporting systems resulting from the normal or abnormal operation, or a failure of any component in the systems.

The interferences between the protective and control systems and the other systems shall be prevented by the absence of interconnections or by use of an appropriate functional separation. If common signals for protection systems and for control systems are used, the respective physical separation shall be provided, e.g. by eliminating interconnection, and it shall be proved that all safety requirements for protective systems are met.

The principles of physical separation shall be used in the design to the most extensive and reasonably feasible degree in order to achieve the guarantee of the maximum independence, namely in relation to the common cause failures. Those principles include:

- separation by distance, layout, orientation, etc.
- separation using barriers
- separation by combining the above-specified principles

The design shall adopt measures providing functional independence and diversity where the systems based on the software means are used, including the human-machine interface, in order to ensure protection against a common cause failure in the cases with high reliability requirements for functionality.

The power plant shall be equipped with a system of unit protection and supporting systems to mitigate the effects of operating states or accident conditions and to ensure that the probability of any release of radioactive substances is acceptably low.

The amount and rate of introduction of reactivity shall be limited in order to ensure that in the case of the reactor protection system activation the reactivity failure (e.g. control cluster ejection) does not result in the damage of the fuel cladding integrity, or pressure boundary of the reactor coolant system or the capability to remove heat adequately from the reactor core.

The protective system of the reactor shall provide initiation of reactor trip system and if necessary the safety system activation in order to maintain the fuel cell integrity within the abnormal state conditions.

The design shall prevent interference among the items important to safety and

it shall ensure that a potential failure of the equipment classified in the lower safety class is not be transferred to the equipment classified in the higher safety class. A failure of systems designed for normal operation shall not have a negative effect on the performance of safety functions.

General requirements related to the safety function performance on various levels shall include:

- necessity to apply the single failure criterion
- provision of the emergency power sources
- necessity of physical separation between the redundant branches in the system
- requirements for automatic control

When using the components or systems providing diversity, it shall be ensured to the reasonably achievable degree that such solution is a safety benefit even when considering the disadvantages including the additional complications during operation, maintenance and testing.

In order to satisfy the safety and operating goals of the power plant, each individual system and group of systems shall be designed using the appropriate degree of redundancy, functional independence, functional and physical separation and suitable cross interconnection. In order to perform a specific function, the systems shall work together to ensure the continuity, redundancy and support of the function execution.

The design and the operation shall establish the procedures of ageing management to confirm that throughout its lifetime, the equipment is capable of meeting the requirements for the performance of safety functions under the conditions of the external environment (e.g. vibrations, temperature, pressure, radiation, humidity) existing before or at the moment when the equipment is required. When specifying those conditions, the effects of various factors affecting the environment, e.g. ageing, that may affect the equipment adversely during its required lifetime shall be considered.

For each system and component, the operating conditions corresponding to the design conditions shall be specified in all relevant categories. The design loads and their combinations shall be taken into account in order to specify the operating conditions of each piece of equipment. If two systems working with the fluid under various pressures are interconnected, then either both systems shall be designed for higher pressure, and/or specific measures shall be taken to prevent exceeding of the design pressure for the system operating with lower pressure.

It shall be proved that the stress corresponding to the design conditions is in compliance with the adopted requirements and standards, e.g. RCCM or Section III ASME Code to ensure the integrity of pressure vessels, pipelines, pumps and valves.

The systems performing the supporting functions such as supplying electric power, cooling, heating, ventilation and air-conditioning shall be classified on the same level as the equipment performing the safety functions since those systems are necessary for the functioning of that equipment.

Reliability, redundancy, diversity and independence of supporting systems, measures for their separation and for testing their operability shall comply with the safety significance of the supported system.

3.3.1.2.12 Criterion of a Single Failure, a Common Cause Failure in the Preliminary Design Concept

The design solution shall be designed with a sufficient degree of resistance against a failure of the items important to safety as a result of a single failure and against common-cause failures.

The systems performing safety functions shall be implemented with the sufficient degree of redundancy using a range of deterministic principles and especially the single failure criterion.

The assembly of equipment meets the single failure criterion if it is capable to perform its safety functions despite a single random failure that is expected to occur in any part of the assembly during any design state in which the respective assembly is required to function. That also includes the hidden (undetected) previous failures. The subsequent failures resulting from the expected single failure are considered an integral part of the single failure.

The assembly of the equipment is defined as a combination of systems and components that perform the specific function. Therefore the required degree of redundancy need not be applied to a single system if another system is available to perform the same safety functions with the performance compliant with the respective safety goals.

When the single failure criteria are applied, the failures of active components are normally considered. The single failures of passive components shall also be considered if it is impossible to prove, with high reliability, that the occurrence of such failure is very improbable and the component functioning shall not be affected by a postulated initiating event. Erroneous action of the system shall also be considered as one of the potential failures.

The single failure criterion shall be applied to all safety systems so that the limits applied in the design basis for such event are not exceeded even on condition of such failure.

If it is necessary to operate several systems simultaneously or consequently in order to perform a specific safety function, the single failure shall be postulated in succession in all systems, however, never in more than one of them at the same time.

The equipment and its components may also be put out of operation due to repairs, periodical maintenance or inspections. During such time-limited period, it is not necessary to apply the single failure criterion for the respective system if it is proved that the combined frequency of the postulated initiating event and losses, or limitation of the safety function will be negligibly low.

The assessment of the required degree of redundancy shall take account of the requirements of the single failure criterion and PSA results.

Where the single failure criterion is applied to safety systems, the redundant equipment shall be provided in compliance with the N+1 concept. Its safety function shall be provided considering the most limiting single failure which may occur in addition to the failure that caused the accident.

Sufficient redundancy shall be taken into account in order to meet the probabilistic goals of the power plant.

If the preventive maintenance is considered for certain equipment providing the safety functions, the N+2 concept (unavailability due to maintenance plus the single failure criterion) shall be applied from case to case depending on the level of the function performed and on the design conditions under which the function may be required.

A failure of multiple equipment or components may occur as a result of a single specific event or cause, i.e. due to a common cause. Such cause may include an imperfection in the design, a manufacturing defect, an error in operation or maintenance, a natural event, a human-initiated event, a signal overload, a change of ambient conditions or an accidental domino effect caused by another power plant operation or failure. The design shall adopt the maximum degree of the reasonably practicable measures in order to minimize such events.

The potential causes of common-cause failures shall be examined in order to determine the requirements of independence, physical separation and diversity.

3.3.1.2.13 Equipment Qualification in the Preliminary Design Concept

Safety systems shall be designed in order to meet, with respect to the risks, the requirements for the required degree of redundancy, diversity and qualification.

It shall be provided for the safety equipment in normal and abnormal operation and in accident conditions that its operability is not limited as a result of a failure of other pieces of equipment located inside the nuclear installation. The safety equipment shall be able to withstand the changes of environment thanks to proper location and adequate protection against dynamic and other effects (flying objects, line vibrations, fluid leakage, overpressure overload), unless it is proved that such events and their effects are practically eliminated.

The design shall ensure that the mechanical and electrical equipment of the safety systems is qualified for use in the operating environment where it is to perform its design function. The qualifications of the mechanical equipment shall meet the applicable industrial standards for the class of the respective equipment. Electrical equipment of safety systems shall be qualified for the environment in accordance with the IEC 780 codes or other standards, on condition that their equivalency to the IEC 780 code for the respective equipment class is proved.

The design shall specify the scheme to define the procedures for maintenance, supervision, periodical testing and replacement of any part to maintain the seismic qualification and the ambient environment qualification for the equipment throughout the power plant lifetime.

In case of the equipment performing its function during the design extension conditions it shall be proved that its functional capability (the so-called endurance) is maintained. The equipment shall be subject to qualification tests to prove the adequate degree of reliability for work in the respective ambient environment for the required period of time.

The procedures to maintain the required equipment qualifications shall be established in order to confirm that the equipment is able to fulfil the requirements for performing safety functions throughout its lifetime if exposed to the ambient conditions (e.g. vibrations, temperature, pressure, radiation, humidity) existing before or at the moment of their need. The qualification scheme shall consider the effects of such environmental factors, including ageing, that may affect the equipment adversely during its required lifetime considering potential degradation as a result of expected corrosion, erosion, material fatigue, etc. Increased attention shall be paid to the qualification of new technologies and equipment.

3.3.1.2.14 Requirements for Monitoring, Testing, Maintenance and Inspections in the Preliminary Design Concept

The design shall specify the operating and maintenance procedures. Those procedures shall include technical specifications, regulations for operation, testing, maintenance, etc. The scope of periodical operating tests and the scheme for performing the periodical tests shall be specified. Within the scheme, the frequency of testing and procedures to determine the increase or decrease of test frequency shall be designed.

The procedures for testing of their basic characteristics shall be specified for individual systems in compliance with relevant codes and standards as specified in Section 1.5, including the relevant equipment and test connections.

Items important to safety shall be designed in order to provide calibration, testing, maintenance, repairs and inspections or monitoring with respect to their functional capability throughout the whole power plant lifetime in all conceivable design conditions based on the design basis in compliance with quality standards corresponding to the importance of safety functions of those items. The design shall enable performing those activities to the extent corresponding to the importance of the safety functions to be performed without exposing the power plant staff to inadmissible exposure. The structure and configuration of the components shall enable inspection during operation. The design of the equipment shall include the measures limiting the occurrence of potential undetected damage during the nuclear installation operation and after postulated initiating events.

The power plant design shall provide performing the periodical availability/operating tests during the power operation, without interrupting the normal operation except for the cases considered counter-productive. Those tests shall not impair the required reliability of the safety function performance. The tests shall focus mainly on continually performed diagnostics of the state of components, their periodical testing, including the test of redundancy signals. Only the tests that cannot be performed during the power operation shall be performed during shutdowns.

If the systems, structures and components classified as safety are not designed in order to provide testing, inspections or monitoring in the required extent, the safety measures compensating the risks of undetected failures shall be adopted. They include the following:

- use of other proven alternative or indirect methods including the reference sample checks and/or use of verified and validated calculation methods
- use of a conservative approach using the safety margins and/or other suitable measures established in order to provide compensation of unexpected failures

The frequency of required inspections and testing shall be specified in order to be sufficient to verify the reliability, however, at the same time it must not result in the excessive degradation of the equipment and unreasonable lapse of lifetime.

The measures providing the checks of equipment availability after maintenance or tests shall be specified. The design shall provide on-line equipment tests while minimizing the risks of interference with safety due to complicated procedures.

Methods of performing the inspections to be used in production shall be compatible with those used for the operating checks and inspections. That way the reference basis for all inspections performed during operation will be set.

The design shall take account of the measures providing the inspections to be performed in all power plant spaces. That includes the requirements for the access to the inspected equipment for the operating staff including the required tools and jigs needed for the inspections.

PSA and risk assessment methods shall be used as the tool for operating inspection optimisation.

The frequency and time required to perform routine inspections shall be optimised, without unfavourable impacts on the reactor’s operation and safety.

3.3.1.2.15 Design Requirements for Unit External Links in the Preliminary Design Concept

Some equipment, e.g. the radioactive waste processing facility, can be shared (common) by several reactor units. If the equipment is shared, especially in the case of safety systems, such sharing shall be specially justified from case to case. The design shall consider the possibility of the arising of accident conditions within several units; the capability to shut down and cool all units must be preserved.

Self-contained systems functioning on the basis of compressed gases, batteries, etc. shall be preferentially used as supporting systems. In principle, the supporting systems may be shared among units in order to increase the safety, but such solution must not result in the increase of probability or extent of impacts of the accident in any of the units.

If the central heating system is designed as part of the design, the measures shall be adopted to prevent the transport of radioactive substances from the power plant to the heating system during all design basis as well as extension conditions.

The unit shall be able to handle the expected grid failures, including the short circuits, fluctuation of grid parameters, voltage drops, voltage collapse and failures of the line without impacts on the functioning of the items important to safety. The effects of frequency and power fluctuations on the operation and reliability of electrical machines and components shall also be taken into consideration.

3.3.1.2.16 Escape Routes in the Preliminary Design Concept

Escape routes shall be designed in compliance with the fire protection standards applicable in the country of the origin of the reactor technology (or applicable in the EU country where the reactor is licensed) and in compliance with Czech fire protection regulations by applying the more stringent of the requirements. The width of escape routes and number of escape exits shall be set depending of the number of employees. The existence of lifts shall not be a reason for decreasing the escape route number or width. Emergency lighting and fluorescent signals shall be provided in the escape routes.

The design shall specify the access routes used by the staff in accident conditions, e.g. to cross from the main control room to the supplementary control room, to the technical support centre and the power plant emergency power supply. In all operating states as well as accident conditions, at least one escape route from all service spaces shall be preserved.

3.3.1.2.17 Means of Communication in the Preliminary Design Concept

The communication equipment shall be designed in accordance with the relevant ambient parameters during its use. Systems of verbal communication shall be designed as support for all phases of operation and maintenance including operation in accident conditions.

Those verbal communication systems shall include:

- means of communication between operators and maintenance and service staff
- information system to inform the staff on the entire premises of the NPP. The range of the system shall involve all required areas of the power plant. Other

visual and acoustic means for notifying the staff in the noisy areas shall be provided:

- mobile radio system (FM), providing radio communication within the range of the entire power plant among the operators, operating staff and maintenance staff
- telephone communication system providing communication among the operators, operating staff and maintenance staff
- plant radio broadcasting covering the entire power plant with voice broadcast
- specialized communication link to external organizations and facilities, including the security bodies, to fire and security authorities, emergency centres outside the power plant location, etc.

Other communication equipment shall be provided in the main control room or its vicinity, including the TV system to enable monitoring the maintenance activities outside the main control room or on the main equipment (as minimum, the state of the operating floor of the reactor, main circulating pumps, strategic points inside the containment, turbo-generator state, etc.).

The main control room shall be equipped with communication equipment to facilitate safe and efficient power plant operation. Special attention shall be paid to the design of communication equipment to be used in the conditions of accidents/failures in communication with the technical support centre. The main control room shall be designed to be used as the power plant communication centre during normal operation and the early phase of an accident.

The communication systems and equipment shall be provided to support all operating and maintenance activities, including accident management. This communication system shall encompass all power plant areas. No dead zones shall be permitted.

The design shall provide the relevant means to support the activities in all operating states as well as accident conditions, including e.g. the technical support centre, gathering place, emergency routes and exits as well as the measures for emergency communication and areas for emergency equipment and facilities.

The system to provide interface and connection with support centres outside the power plant shall be available to enable the data exchange with the power plant.

As required, several support centres can be expected.

The communication equipment design shall enable regular testing.

3.3.1.2.18 Human Factor in the Preliminary Design Concept

The design shall take account of the issues of human factor and human-machine interface in compliance with good ergonomic principles. The design solution shall support successful activity of the staff in terms of the intervention time available, expected work environment and psychological stress for all considered power plant states. The design shall take account of the fact that the reliability of staff intervention can only be ensured if the staff has enough time to make decisions and to perform an action if the required information is provided in a comprehensive, easily processible and unambiguous form and if the work environment after the event is acceptable.

The same criteria and philosophy considering the importance of human-machine interface shall be applied for all power plant systems.
The design shall support the success of the operator's intervention in relation to the time available. The requirements for immediate operating staff interventions shall be minimized.
The design shall respect proper ergonomic principles for the operator's workplace and minimizing loads in operating and maintenance activities. In order to achieve that, the design shall consider sufficient margins, extend the use of simplified systems, and perform the optimisation in order to enable natural behaviour aimed to establish modern and reliable human-machine interface.

The design shall include the requirements for operating and maintenance staff including its design of organization and number of employees required for individual operating and maintenance activities of the power plant and the required staff qualification. The minimum number of the operating staff necessary to put the power plant into a safe condition shall be specified.

The principles concerning the human factor shall be consistently applied during the entire design for each operating or maintenance activity in order to reduce errors in operation and maintenance during all power plant operational modes.

The tasks for equipment maintenance shall be specified, including defining of requirements for expertise, tools, test equipment, access, environmental conditions, etc. These tasks shall also include all tests necessary for re-commissioning of the equipment after the maintenance.

In ergonomic terms the basic requirements for the presentation of information on the basis of VDU (Video-Display Unit) shall be almost the same as in the case of conventional control panels. For that reason, most of the standards that apply to the conventional human-machine interface shall also apply to the digital human-machine interface. In addition to the conventional presentation, the computer human-machine interface has other beneficial properties including:

- location of the relevant information using an easy and self-explaining navigation via the display navigation
- use of warning signals in the plain text
- possibility to increase the “intelligence” while handling and sorting the failure signals and other process information

The following principles shall be applied:

- only the information required for the tasks of the operator enabling the prompt assessment of the state and specifying the required manual interventions shall be displayed. All other information shall be removed
- visual and audio indications shall be provided to notify the operating staff of the deviations from the operational states and processes having impacts on the nuclear safety
- the information shall be structured in order to facilitate the operator to search for the necessary information on various hierarchical levels in the system or to switch among displays
- the operator shall have access to more than one VDU at the same time. That assumes the use of several VDUs at one workplace

- the related information shall be grouped
- VDU presentation shall be compatible with other forms of presentation in the power plant
- density of information on less frequently used displays shall be lower than the density of information on more frequently used displays
- the colour shall not be the only carrier of the information
- the number of graphical resolution shall be limited

the information and control devices shall be designed in order to enable the operators to perform their tasks correctly, quickly and safely and they shall enable the safe operation of the power plant within the limit parameters for individual systems.

The information and control elements shall be logically/functionally grouped using the same way as the elements providing the feedback for control and interventions on the state panels of the equipment and/or process (the same format or the same panel element). The operator shall be able to obtain the additional updated information on the power plant state any time from the information and control equipment (grouping of logical schemes in case of the conventional panel, adapted formats for the computer control).

The power plant shall be designed in order to meet the following autonomous goals:

The limit values for releases in all operational states as well as accident conditions shall be observed even without an intervention of the operator from the main control room for at least 30 minutes from the start of the accident and in the absence of any intervention outside the control room for at least 1 hour (from the start of the accident).

The power plant design shall apply the specific requirements important to the reliability of the human factor including the provision of adequate space, lighting, noise level and work environment. All power plant spaces shall provide a suitable environment providing safety and comfort for the power plant personnel and for the availability of the power plant equipment during the normal and abnormal operational conditions and in accident conditions.

The main control room shall be located in order to enable convenient operational control in all power plant conditions. The internal and external risks shall also be considered, including projectiles, radiation, fire, etc. The design shall adopt the measures to maintain the availability of the control room to enable power plant monitoring also during failures and accidents.

For the scope of defined operational states, the relevant control system analysis shall be performed (e.g. using simulation). In the process of designing I&C, the human factor shall be fully integrated, considered and documented. Mock-ups, simulations or experiments shall be used in the early phase of designing to assess the specific design characteristics and solutions and to support the interactive development. The plans of verification and validation (V&V) shall be developed. The V&V plan development shall be based on relevant standards. For those purposes the full-scale simulator shall be used in the earliest phase possible.

3.3.1.2.19 Ageing Management in the Preliminary Design Concept

The procedures to confirm that throughout its lifetime the equipment is capable of meeting the requirements for the performance of safety functions in the ambient conditions (e.g. vibrations, temperature, pressure, radiation, humidity) existing before or at the moment when its function is needed shall be established. The effects of other environmental factors, e.g. ageing, that may affect the equipment adversely during its required lifetime shall be taken into consideration. The safety margins shall also take account of the uncertainties of the initial state setting and the uncertainties of assessment methods.

The design shall provide a list categorizing the items (e.g. components and subsystems) based on their potential design lifetime. The scope and frequency of maintenance and inspection activities for each item shall be incorporated in the planning of maintenance and inspection schemes starting at the stage of construction. Maintenance and inspection activities for each item shall comply with the design lifetime of the respective item.

The information for individual items shall include:

- expected lifetime of the component
- limiting conditions of the environment that the assumption of the lifetime is based on
- methodology used to specify the lifetime penalization in the case that the limiting environmental conditions for the lifetime are exceeded
- recommended maintenance required to comply with the expected lifetime
- and recommended spare parts for the components necessary to maintain the expected readiness of components and systems

The ageing management scheme shall be established in order to monitor all ageing mechanisms of the items important to safety, to specify potential effects of ageing and to specify the measures to provide availability and reliability of the equipment. The specific measures preventing the known, long-term failure mechanisms known from the operation of existing operated power plants shall be implemented. The power plant shall be designed so that the known failure mechanisms do not prevent the power plant from achieving its design lifetime or complying with the requirements for readiness and reliability.

At least the following mechanisms shall be taken into consideration:

- embrittlement of carbon steel structures
- cyclic fatigue due to vibrations, heat expansion, etc.
- both even and uneven corrosion (e.g. caustic embrittlement, corrosion caused by acids, microbiologically induced corrosion)
- flow accelerated corrosion, cavitation, erosion
- cracking of pumps, valves, turbines, etc.
- corrosion of tubes and shells of steam generators, feed water heaters and condensers
- strain ageing of fans, relays and other components

- heat ageing of electrical components
- fatigue of connectors and non-metal components
- electrolyte failure
- radiation ageing of electrical components in the core and its vicinity
- loss of strength and/or fracture resistance due to the change of metallurgical state of alloys resulting from the exposure to high temperatures or thermal cycles and/or high radiation

In addition to the above-specified, the systems, structures and components shall be designed in order to enable calibration, testing, maintenance, repairs and inspections or monitoring in compliance with the qualification scheme with respect to their functional capability throughout the entire power plant lifetime.

The qualification scheme, plan for achieving the power plant design lifetime and maintenance schemes shall include the monitoring of the state of all structures, systems and components where such monitoring is cost efficient in terms of the safety, reliability and maintenance costs and shall specify the respective tests before the commissioning as well as during the operation proper, also considering the performed maintenance, modifications and testing.

The cumulative scheme shall be developed in order to obtain the data to assess the actual component lifetime depending on their operating history and measurement of the characteristics affecting their lifetime so that the sufficient and suitable monitoring and recording systems are available.

The required design lifetime of non-replaceable structures, systems and components shall be at least 60 years.

The ageing management scheme shall identify all systems, structures and components that do not comply with the previous requirement (60-year lifetime for non-replaceable structures, systems and components) and specify the need for spare parts.

The power plant design shall be optimised in order to meet the power plant lifetime requirements. The optimisation shall take into account the following:

- all main power plant components shall be required to comply with the required design lifetime
- the power plant design shall enable the replacement of all operating components and equipment including the heavy components, except the reactor pressure vessel
- Within the reactor pressure vessel ageing management, all relevant factors including the embrittlement, thermal ageing and wear shall be taken into consideration in order to compare the actual state with the prediction throughout the whole lifetime
- In order to support the achievement of the ETE3,4 lifetime level of 60 years, the design shall propose the strategy to include:
    - identification, assessment and inclusion of all mechanisms of potential ageing of structures, systems and components

- substantiating methods of the ageing effect assessment shall be defined for each element and each mechanism of degradation in order to confirm the compliance with the design lifetime
- review with respect to the respective experience and research results
- consideration of the use of modern materials with higher ageing resistance
- development of preventive maintenance schemes to use the monitoring of structures, systems and components, provided it is effective in terms of costs
- optimization of frequency and effects of transients
- provision of adequate monitoring systems for monitoring and recording of significant processes during the design lifetime
- location of the components leading to elimination or minimizing of the frequency of exposure in an unfavourable or lifetime shortening environment
- launch of the system of lifelong records to contain the provisions of the information relevant for the design lifetime assessment including the input parameters of the components, environmental limits, acceptance criteria and computer databases of related parameters

3.3.1.2.20 Measures for Construction and Decommissioning in the Preliminary Design Concept

Achieving the design requirements and required safety characteristics of the items important to safety in manufacture, designing, assembly, installation and construction shall be, among others, provided in the way of specifying the regulations for the respective activities to be used in designing, manufacture and construction complying with the safety categorization of structures, systems and components.

The design shall utilize the proven technologies that were successfully used in the existing pressurized water reactors or in the analogical working conditions and applications in other industries (at least for several years) and it shall be proved that the experience is appropriate for the specific nuclear utilization.

The equipment shall be designed in order to facilitate the access of personnel to perform inspections, maintenance, repairs, decontamination, replacement of redundant parts or disassembly and thus provide the reduction of radiation exposure of the operator’s personnel. The requirements for the implementation of the processes shall already be reflected at the designing stage.

A feasibility study shall be developed proving that the ETE3,4 design provides implementation of the future decommissioning strategy including the estimate of decommissioning costs aimed to provide the operator with sufficient substantiation for the specification of the amount of payments to the decommissioning funds. The study shall contain:

- proof that decommissioning based on the designed plan of ETE3,4 installation and measures for maintenance and repairs is practicable and shall be performed safely with regard to the significance of the employee health and safety, protection of public and environment and in compliance with applicable codes and standards (mainly the Decree No.185/2003 Coll. of the State Office for Nuclear Safety) using engineering techniques available at the time of commissioning (e.g. robotics, decontamination equipment and special devices), and with regard to applying the ALARA principle

- appreciation of the activity inventory, assessment of the expected types, sources and volumes of waste (inactive as well as radioactive), including the hazardous (inactive) waste

The design shall provide the summary of design measures facilitating the decommissioning. Mainly the following elements to be used in the design are important:

- design simplification – reduced number and complexity of systems, minimum inter-dependency of systems, simple isolation of systems
- availability of the equipment and spaces required for handling and storage of radioactive waste
- simple disassembly and removal of the reactor pressure vessel and other large components of the primary coolant circuit
- design of biological screening The design shall consider the fact that biological shielding will have to be disassembled. The reactor biological shielding shall be designed and manufactured from detachable modules
- the selection of materials aimed to reduce the risks following from the dose rate in the radioactive material vicinity and reduction of the amount of radioisotopes with long half-life
- minimizing the possibility of activation of structures, systems and components in the reactor pressure vessel vicinity and minimizing the extent of contamination within the whole ETE3,4
- achieving the proper finish of the potentially contaminable metallic components to facilitate the subsequent potential decontamination:
    - to minimize the amount of waste to be stored
    - to enable the required handling
    - easy availability and easy removal of components and parts in order to:
        - minimize the radiation dose of the personnel
        - reduce the number of employees required for decommissioning work
          o selection of design and geometry of structural elements in order to minimize the pollution in the areas where it can accumulate

3.3.1.2.21 Physical Power Plant Protection in the Preliminary Design Concept

The design solution of the physical protection shall be designed in order to provide the required way of physical protection of the nuclear installation and nuclear material.

Physical protection of the power plant shall fully respect the Analysis of Possibilities and Needs for Physical Protection drawn up by the course of provisions of Section 13

(3) (d) of the Act No. 18/1997 Coll. [L. 2] as a part of application for licence and the content of this separate documentation shall correspond to the requirements of Appendix A. II. of that Act. Based on the preliminary assessment of the risks following from unauthorized activities with nuclear material (NM) and nuclear installation (NI) and expected proposal to include the NM and NI in the respective category in terms of physical protection, the preliminary proposal of technical and administrative measures for physical protection was developed in compliance with the requirements of Decree No. 144/1997 Coll. [L. 285], which also respect the latest recommendations of INFCIRC/225/Rev. 5.

Partial Preliminary Assessment

The preliminary design concept summarizing the most important requirements for the nuclear power plant design, its safety and technological functions specified in Sections 3.3.1.2.1 to 3.3.1.2.21 “Basic Requirements to Provide Safety in the Preliminary Design Assessment” creates the preconditions to comply with the requirements of the Decree No. 195/1999 Coll. [L. 266] Sections 3, 4.1, 4.2, 4.3, 7, 11,12, 18.1, 18.2, 19.1 and 2, SÚJB Safety Guide BN-JB-1.0 [L. 276] (1-5, 9, 10, 10.1-10.3, 11-14, 19, 20-25, 27-28, 31, 33, 35, 37, 39, 41-45, 48-51, 95, 127, 136, 143, 144), IAEA SSR 2/1 [L. 252] 2.6-2.14, 4.1-20, 5.15.7, 5.10, 5.24, 5.25, 5.31, 5.32, 5.33, 5.37, 5.39, 5.40, 5.44-74, Req. 4, 5, 6, 7, 8, 10, 11, 12, 13, 14, 15, 16, 20, 21, 23-27, 42 , WENRA [L. 27] App. E 1.1, 2.1, 2.2, 3.1, 4.1, 4.2, 5.1, 6.1, 7.1-5, 8.2, 9.1-4, 10.7, 10.8, 11.1 Issue I 1.1, 2.1, 2.2, 2.4, 3.1, 3.2, Issue G 4.1, 4.2, Issue S 2.5, Issue R 4.2, 4.4, WENRA NEW [L. 270] 01-06.

3.3.1.3 SUMMARY PRELIMINARY EVALUATION OF THE DESIGN CONCEPT IN THE AREA OF MEETING THE DESIGN REQUIREMENTS FOR STRUCTURES, COMPONENTS, SYSTEMS AND EQUIPMENT

The ETE3,4 design shall meet the safety requirements specified in Section „3.3.1 Compliance with Basic Requirements of State Supervision in Order to Provide Safety" and at the same time specified in Sections 3.3.(2-13), which are based on the requirements stipulated by the Act No. 18/1997 Coll. [L. 2] and its implementing decrees for nuclear safety, radiation protection and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270].

To meet those requirements, the ETE3,4 shall use systems, structures and components and their equipment, which shall ensure (with adequate reliability and resistance) the technological functions relevant in the qualitative and quantitative way in accordance with the specified requirements for the prescribed safety functions.

The principles of the design solutions characterized in Section 3.3.1.2 Basic Requirements to Provide Safety in the Preliminary Design Assessment were developed based on the requirements submitted by the licence applicant applied to potential suppliers of the nuclear installation within the tender, and they make up the concept of the design solution for this part of the design. The partial assessments provide evidence that the expected design will meet the specific requirements for systems, structures and components and safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], SÚJB Safety Guide BN-JB-1.0 [L. 276], document IAEA SSR 2/1 [L. 252], document WENRA [L. 27] and WENRA NEW [L. 270]. The systems, structures and components and equipment comply with the performed preliminary assessment of the design concept and create the precondition to fulfil the defined scope of basic safety rules and principles in the design of the future new nuclear installation in the Temelín location.

The particular method of implementation of the individual safety requirements specified in Section 3.3.1.2 shall be specified in detail in the nuclear power plant design.

3.3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

The section contains the basic legislative requirements for the classification of systems, structures and components (equipment), which is performed in order to provide their categorization to safety classes based on the importance to safety. The requirements, rules and specifications shall be defined for individual safety classes in order to provide the maximum reasonably achievable level of nuclear safety and reliability. At the next stage of licence documentation, this section shall include the seismic classification of systems, structures and components and classification of systems based on the quality groups to a depth of detail and in the structure set out in RG 1.206 [L. 275].

3.3.2.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE DESIGN OF STRUCTURES, COMPONENTS, SYSTEMS AND EQUIPMENT

3.3.2.1.1 Purpose and Importance

Reference: SÚJB Safety Guide BN-JB-1.0 (6), WENRA Issue G 1.1, 3.1, IAEA SSR 2/1 Req. 23, 5.37

Each item important to safety (selected equipment), including both hardware and software, shall be identified and classified (i.e. included into categories – safety classes) based on the significance of the safety function provided.

The design of the items important to safety shall provide that the items are qualified, obtained, installed, commissioned, operated and maintained in order to be able to withstand, with sufficient reliability and efficiency, all conditions specified in their design basis.

Reference: SÚJB Safety Guide BN-JB-1.0 (45), IAEA SSR 2/1 Req. 22, 5.35, 5.36, Req. 27, 5.42, 5.43 WENRA Issue G 3.2

The design shall eliminate interference among the items important to safety and it shall ensure that a potential failure or fault of the equipment classified in the lower safety class does not propagate on the equipment classified in the higher safety class.

The equipment performing multiple functions shall be classified according to the most important function performed in terms of the safety.

Auxiliary systems providing the availability of items important to safety shall be classified similarly. Reliability, redundancy, diversity and independence of auxiliary systems and measures for their isolation and testing of their functionality shall be adequate to the importance to the safety of the system they support. It will be inadmissible for a failure of an auxiliary system to simultaneously affect the redundant parts of a safety system or a system performing diverse safety functions and to endanger the capability of the system to perform its safety functions.

Failures of systems designed for normal operation shall not affect performance of safety functions.

Reference: IAEA SSR 2/1 Req. 23, 5.38

Reliability of an item important to safety shall comply with its safety significance.

In the selection of the equipment, both ways of its failure and its potential incorrect functioning shall be considered. The design shall prefer equipment that has predictable and apparent ways of failure and with the design facilitating repairs or replacement.

Reference: WENRA Issue G , 4.1, 4.2

The design or the items and used materials important to safety shall consider the effect of the environment and design basis accidents on their properties and performance throughout the whole lifetime of the nuclear power plant.

Such procedure for the equipment qualification shall be adopted in order to confirm that during their design lifetime the items important to safety shall fulfill the requirements imposed on their functionality.

The adopted procedure for equipment qualification shall also consider:

- effects of the environment where the equipment is located during its lifetime
- whether the functionality of the equipment is required in case of the rise of expected operational events and in accident conditions

3.3.2.1.2 Classification Method

Reference: SÚJB Safety Guide BN-JB-1.0 (7, 8, 122), IAEA SSR 2/1 Req. 22, 5.34, WENRA Issue G 2.1

The safety criticality classification of a safety function shall be based on the deterministic approach supplemented, in proper cases, with the probabilistic approach and engineering judgement while considering the following factors:

- the safety function that is provided by the equipment
- impacts of the safety function failure
- frequency of the required safety function performance
- how long after the arising of a postulated initiating event and how long after initiation the equipment is supposed act

The safety class classification criteria for selected equipment are specified in a special decree (Decree No. 132/2008 Coll. [L. 258]).

The selected equipment, a potential failure of which can cause, directly or indirectly, the release of radioactive substances or ionising radiation and/or otherwise threaten nuclear safety and human health, forms the category of "specially designed selected equipment". Methods of specification of this equipment and technical requirements for their technical safety are stipulated by the special decree (Decree No. 309/2005 Coll. [L. 280]).

3.3.2.1.3 Safety Class Requirements

Reference: WENRA Issue G 2.2

Based on the classification, the following shall be specified for each safety class:

- appropriate codes and standards for the design, manufacture, installation and inspections
- need for redundant power supply and qualification for ambient conditions
- states of availability and unavailability of the systems providing safety functions to be considered in the deterministic safety analyses
- respective quality and reliability requirements

3.3.2.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS IN THE PRELIMINARY DESIGN CONCEPT

This section contains the characteristics of the system of classification of structures, components and systems in the preliminary design concept. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant.

During designing, manufacture, implementation and operation, the functional classification and categorization of the equipment shall contribute to provision of the quality of the systems and items important to safety.

Basic Requirements for the Classification of Structures, Components and Systems

The design shall propose, describe and apply the appropriate and consistent system of classification and categorization including the associated safety requirements like seismic classification and categorization, qualification for the environment, requirements for design standards and requirements for quality assurance.

The design shall prove the appropriateness of the proposed classification and categorization system with respect to licence requirements.

Design loads shall be specified in compliance with the classification, construction methods, other available information and applied codes and standards.

A failure of structures, systems and components in one safety class shall not cause a failure of other structures, systems and components in a higher safety class.

Classification Method

Within the design, the structures, systems and components shall be classified into safety classes in compliance with the respective Czech legislation, namely the Decrees No. 132/2008 Coll. [L. 258] and 309/2005 Coll. [L. 280].

Categorization of relevant equipment shall be performed using the deterministic approach complemented by probabilistic methods, as needed.

The classification and categorization shall be performed on the following basis:

- defining of safety functions required to achieve and maintain the controlled state or safe shutdown state after a severe accident
- identification of structures, systems and components for each function

- classification of all structures, systems and components to safety categories based on the highest safety function performed
- classification of all structures, systems and components into the classes based on the codes and standards applied for the design

The classification and categorization system shall take account of the fact that some systems, structures and components traditionally classified as unimportant to safety can substantially contribute to safety since they form another level within the defence in depth. For that reason, especially in the design maximizing the use of passive safety systems, the importance of such structures, systems and components shall be assessed and special treatment designed for those structures, systems and components.

Safety Class Requirements

For the systems, structures and components of nuclear installations included in the highest safety category, the proper group of codes and standards for designing and construction shall be specified to be used in their design and construction.

A tiered approach shall be used in categorizing the selected equipment so that class one includes the selected equipment subject to the highest requirements for reliability, qualification, quality assurance, number and scope of inspections, and the related documentation.

Specified requirements shall be defined for each category in respect of:

- codes and standards used in designing, manufacture, construction
- requirements for provision of power supply
- qualification for external work conditions
- seismic classification
- availability in case of the rise of an initiating event that can be assumed in deterministic analyses
- quality assurance
- operating inspections
- periodical tests
- requirements for equipment reliability

Specifically, the requirements shall be defined for the equipment and systems designed to handle severe accidents.

Partial Preliminary Assessment

The preliminary design concept summarizing the most critical requirements for systems, structures and components specified in Section "3.3.2.2 Classification of Structures, Components and Systems" creates preconditions for meeting the requirements of the SÚJB Safety Guide BN-JB-1.0 [L. 276] (6, 7, 8, 45, 122), IAEA SSR 2/1 [L. 252] (Req. 22, 23, 27), WENRA [L. 27] Issue G (1.1, 2.1, 2.2, 3.1, 3.2, 4.1, 4.2).

3.3.3 PROTECTION AGAINST EXTERNAL EFFECTS

3.3.4 PROTECTION AGAINST EXTERNAL EFFECTS

The section contains the basic legislative requirements related to protection against external effects. Protection against external effects means the design of the nuclear installation resistant against the effects of natural events and human induced events when the performance of fundamental safety functions needs to be ensured. Protection against external effects shall be solved in the design in compliance with the design events and parameters specified in Chapter 2 hereof.

At the next stage of the licensing documentation this section shall contain the proof of design resistance against the load due to wind and tornadoes and the effect of floods at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.3.4.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE PROTECTION AGAINST EXTERNAL EFFECTS

Reference: SÚJB Safety Guide BN-JB-1.0 (23)

The deterministic and probabilistic methods or their combinations shall be used to draw up a list of postulated initiating events that may have a significant effect on the nuclear installation safety including those that may be caused by internal or external effects induced by natural events as well as human activity, and/or a combination of those events.

In order to provide the nuclear power plant protection against the exposure to external effects the following principles shall be applied in the design:

Reference: IAEA SSR 2/1 Req. 17, 5.17 - 5.20, 5.22

- Identification of all predictable external effects, including the human induced events that may affect, directly or indirectly, the nuclear power plant safety including the assessment of their impact on the safety
- The corresponding assessment of natural events and human induced events that were identified in Section 2 hereof
- The influences of external effects shall be used to specify the postulated initiating events and the subsequently developed loads shall be applied in the design of the items important to power plant safety
- The nuclear power plant safety shall not be permitted to be dependent on the availability of off-site services such as electricity supply and fire fighting services. The design shall take due account of site specific conditions to determine the maximum delay time by which off-site services need to be available
- Items important to safety shall be designed and located to minimize, consistent with other safety requirements, the likelihood of and the possible harmful consequences of external events
- Features shall be provided to minimize any interactions between buildings containing items important to safety (including power cabling and control cabling) and any other plant structure as a result of external events considered in the design

- The design shall be such as to ensure that items important to safety are capable of withstanding the effects of external events considered in the design, and if not other features such as passive barriers shall be provided to protect the plant and to ensure that the required safety function will be performed
- The design of the power plant consisting of multiple nuclear units shall consider the potential simultaneous impact of various specific phenomena/events on individual nuclear installations in the respective site

Reference: Decree No. 195/1999 Coll., Article 10, SÚJB Safety Guide BN-JB-1.0 (24), (25)

Nuclear installations shall be designed in order to enable, in the case of the natural events that cannot be practically eliminated (specified in Section 3.3.4.1.1) or the human induced events from the outside of the nuclear installation that cannot be practically eliminated (specified in Section 3.3.4.1.2), specifically to:

- safe shut down the reactor and to keep it in sub-critical state
- to remove the residual power of reactor for a sufficiently long period
- ensure that potential radioactive releases do not exceed the values stipulated by special legal decree (Decree No. 307/2002 Coll. [L. 4])

In the designing process of the nuclear installation, the following shall be considered:

- characteristics of the site where the nuclear installation is to be located, in compliance with the requirements of special legal decree (Decree No. 215/1997 Coll. [L. 1])
- the most significant natural events or human induced events historically recorded in the specific site and its vicinity, extrapolated while taking account of the limited accuracy of values and time
- combination of effects of natural events or human induced events and the states of abnormal operation or accident conditions caused by such events

3.3.4.1.1 Natural Events

Reference: Decree No. 195/1999 Coll., Article 10, SÚJB Safety Guide BN-JB-1.0 (24), IAEA SSR 2/1 Req. 17, 5.17, WENRA App. E 5

The design shall be solved in order to provide the resistance of the items important to safety against natural events, namely:

- Earthquakes
- Floods
- Extreme wind
- Extreme weather conditions
    - Extreme outside temperatures
    - Extreme cooling water temperatures
    - All forms of precipitation
    - Humidity

- Hoarfrost

- Effects of flora and fauna

3.3.4.1.2 Human Induced Events

Reference: Decree No. 195/1999 Coll., Article 10, SÚJB Safety Guide BN-JB-1.0 (24), IAEA SSR 2/1 Req. 17, 5.17 - 5.20, WENRA App. E 5

The design shall be solved in order to provide the resistance of the items important to safety against the human induced events as described in Sections 2.2 and 2.3 hereof, namely:

- Aircraft crash
- Explosions
- Fires
- Other traffic and industrial accidents
- Electromagnetic interference
- Interference by other technical equipment outside the nuclear installation
- Gases affecting the habitability of control rooms

3.3.4.2 PROTECTION AGAINST THE EXTERNAL EFFECTS IN THE PRELIMINARY DESIGN CONCEPT

This section contains the characteristics of the system of protection against the external effects in the preliminary design concept. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant.

General Principles of Protection against External Effects

Technical parameters of the nuclear installation shall be in compliance with the criteria following from the legislative requirements and Section 2 hereof, which specifies the requirements in terms of potential effects of the natural events and effects caused by human activity.

The systems, structures and components, required to maintain the safe state of the nuclear power plant, shall be designed in order to resist all natural events expected in the site location as well as the human induced events.

The design shall consider the most serious natural events historically recorded in the respective site and its vicinity, extrapolated while taking account of the limited accuracy of values and time. The combinations of effects of natural events and the human induced events and accident conditions caused by such events shall further be considered.

The design shall assume the rise of the events in the most unfavourable operating conditions and it shall also consider the secondary effects of the event (e.g. vibrations, induced fires or explosions).

The design shall provide maintaining of the safe state of the nuclear unit and removal of the residual heat for the sufficiently long time in case of the least favourable postulation of the initiating event, using the power plant means (i.e. without the external supply of energies and materials).

Natural Events

Natural events for the purpose of the nuclear installation design mean the outdoor external effects of natural origin in the course of which the performance of fundamental safety functions needs to be provided. The design shall be solved in order to provide, with sufficient margin, the resistance of the nuclear installation against the potential natural disasters following from Chapter 2 hereof. They include, in particular:

- hydrological and hydrogeological conditions
- weather conditions
- geological, geotechnical and seismic conditions
- criteria for nuclear installation siting

Remark: the issues of seismic resistance are elaborated in detail in sections "3.3.7.2 Seismic Resistance in the Preliminary Design Concept", "3.3.8.2 Construction of Seismic Resistance Category I Buildings in the Preliminary Design Concept" and "3.3.10.2 Seismic and Dynamic Resistance of the Mechanical, Electrical and I&C Equipment in the Preliminary Design Concept".

External Effects Caused by Human Activity

The design shall be solved in order to provide, with sufficient margin, the resistance of the nuclear installation against the potential external effects caused by human activity. The analyses specified in Chapter 2 hereof have been prepared for such activities. They mainly include the effect of industrial, transportation and military structures in the site and analyses of risk activities on the power plant premises. Regardless of the low probability of an accidental aircraft crash the power plant design and dimensioning of its structures and systems shall deterministically consider a deliberate aircraft crash, starting from a small sports plane to military and large passenger plane.

Among the potential main negative effects, short-term contamination of the ground atmospheric level in the case of an accident during the transport of chemicals related to their release cannot be eliminated. This risk for the habitability of control rooms shall be eliminated in the design using a proper design solutions of those spaces important for safety.

Partial Preliminary Assessment

The preliminary design concept summarizing the most important requirements for systems, structures and components, safety and technological functions of safety systems specified in Section "3.3.3 Protection against External Effects" creates the preconditions for the fulfilment of the requirements of Decree No. 195/1999 Coll. [L. 266] Article 10, SÚJB Safety Guide BN-JB-1.0 [L. 276] (23, 24, 25), IAEA SSR 2/1 [L. 252] Req. 17, 5.17 - 5.22, WENRA [L. 27] App. E 5.

3.3.5 PROTECTION AGAINST INTERNAL EFFECTS

3.3.6 PROTECTION AGAINST INTERNAL EFFECTS

The section contains basic legislative requirements related to the protection against internal effects. Protection against internal effects means the design of the nuclear installation resistant against the effects of defined events when the performance of fundamental safety functions needs to be provided. It is aimed to provide the functionality of the items important to safety in the normal and abnormal operation, during tests as well as upon the arising of accident conditions.

At the next stage of licensing documentation this section shall contain the proof of design resistance in terms of protection against flying objects, and protection against dynamic effect postulated by the rupture of pipeline at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.3.6.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE PROTECTION AGAINST INTERNAL EFFECTS

Reference: SÚJB Safety Guide BN-JB-1.0 (23)

The deterministic and probabilistic methods or their combinations shall be used to draw up a list of postulated initiating events that may have a significant effect on the nuclear installation safety including those that may be caused by internal or external effects induced by natural events as well as human activity, and/or a combination of those events.

Reference: IAEA SSR 2/1 Req. 17, 5.16

In order to provide the power plant protection against the consequences of internal effects, the design shall provide the identification of all predictable internal effects affecting directly or indirectly the nuclear installation safety including the assessment of potential safety impacts.

The load from the exposure to internal effects shall be applied in the design of the items important to safety.

The design shall consider all internal effects, namely:

- internal fires and explosions
- internal flooding
- internal projectiles
- falling objects
- collapse of supports and other structural parts
- pipe whips
- jet impact
- release of fluid from failed systems or other systems located on the site

Reference: IAEA SSR 2/1 5.16

The design shall incorporate appropriate features for prevention and mitigation shall be provided to ensure that safety is not compromised.

3.3.6.2 PROTECTION AGAINST INTERNAL EFFECTS IN THE PRELIMINARY DESIGN CONCEPT

This section contains the characteristics of the system of protection against the internal effects in the preliminary design concept. The design characteristics were identified based on the technical part of the BIS specifying the requirements for safety and technical design solution of the future power plant

The systems, structures and components essential to maintain the safe state of the power plant shall be designed in order to avoid, during the normal and abnormal operation, testing and upon the rise of accident conditions, the loss of their functioning as a result of failures of other pieces of the equipment located inside the nuclear power plant. The items important to safety shall be capable of preserving their functionality even in case of changes of the environment related to the failures of other equipment. That means that they shall be designed for the worst environment that may occur in the place of their location as a result of accident conditions.

The design shall consider the specific loads and parameters of the environment acting on the equipment, structures and systems as a result of internal effects. The internal effects include the effects of leakage of coolant from high-energy pipeline, pipe whips, internal projectiles, internal flooding, internal fires and explosions, falls and impacts of heavy burdens, failure of pressure parts, braces and other structural parts, electromagnetic interference with the power plant equipment, leakage of water, gas, steam or harmful substances.

Protection of the items important to safety shall be solved using the consistent functional and spatial separation of individual subsystems that will be redundant. Besides, the equipment shall be protected by proper location and mechanical barriers unless it is proved that it resists the considered effects.

Partial Preliminary Assessment

The preliminary design concept summarizing the most important requirements for systems, structures and components, safety and technological functions of safety systems specified in Section “3.3.5 Protection against Internal Effects” creates the preconditions for the fulfilment of the requirements of SÚJB Safety Guide BN-JB-1.0 [L. 276] (23), IAEA SSR 2/1 [L. 252] Req. 17, 5.16.

3.3.7 SEISMIC RESISTANCE

The section contains the basic legislative requirements for the seismic resistance of the design. Seismic resistance represents the summation of the requirements and principles to be applied to the design and construction of the items important to safety, in order to provide the reliable performance of all safety functions in case of a seismic event adequate to the design events and parameters. At the next stage of the licensing documentation, this section shall contain the proof of design seismic resistance at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.3.7.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE SEISMIC RESISTANCE

Reference: IAEA SSR 2/1, Req. 18

The design procedures for civil structures, technological assemblies and power plant equipment shall be specified in compliance with the relevant national and international codes and standards and in compliance with the proven engineering practice, considering the nuclear safety requirements.

Reference: ISAR Section 2.7

The civil structures, technological assemblies and equipment shall be designed in compliance with the requirements contained in Section “3.3.4 Protection against

External Effects” and in compliance with the design events and parameters specified in Section “2.6 Geological, Geotechnical and Seismic Conditions“.

Reference: IAEA SSR 2/1, 5.21

The NPP design shall provide a sufficient safety margin for the protection against the effects of seismic load and prevent sudden serious failure immediately after the design values have been exceeded.

3.3.7.2 SEISMIC RESISTANCE IN THE PRELIMINARY DESIGN CONCEPT

This section contains the characteristics of the system of provision of seismic resistance of the equipment in the preliminary design concept. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant.

Structures and equipment shall be classified into seismic categories based on their safety requirements for the integrity and availability during and after an earthquake. Categorizing shall also ensure that the design and execution standards are selected and applied systematically and consistently. The seismic category of a structure or equipment shall be chosen based on the examination of its safety function during and subsequently after the design basis earthquake, in a way corresponding to the classification of the equipment into safety categories.

The power plant shall be designed in order to provide its reliable and safe shutdown and cooling down even after the maximum design basis earthquake.

The power plant resistance against the maximum design basis earthquake shall be higher than required by the IAEA Safety Guide NS-G-3.3, which specifies the horizontal acceleration component 0.10g.

For the larger than design basis earthquakes an appropriate method shall be used to prove that they do not have any significant consequences.

Within the design, the analysis shall be performed proving that the structures and equipment have adequate safety margins (the so-called SMA analysis, Seismic Margin Assessment). The analysis shall prove that in fact there are sufficient margins in the seismic design of main structures and components after the design conditions have been exceeded. SMA analysis shall prove that the design is able to withstand the earthquake with the horizontal acceleration 40% higher than the design basis earthquake level.

The analysis will be intended to determine, with a high degree of certainty, the seismic resistance of the minimum set of the equipment and structures required to avoid core damage, to bring the plant to and to maintain it in a safe shutdown state.

Partial Preliminary Assessment

The preliminary design concept summarizing the most critical requirements for systems, structures and components specified in Section "3.3.7.2 Seismic Resistance" creates the preconditions for meeting the requirements of IAEA SSR 2/1 [L. 252] (Req. 18, 5.21).

3.3.8 DESIGN OF THE SEISMIC RESISTANCE CATEGORY I STRUCTURES

The section defines the requirements for the design of structures important to maintain the safe state of the nuclear unit during the earthquake and after it subsides. At the next stage of the licensing documentation, this section shall contain the proof of design seismic resistance at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.3.8.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE DESIGN OF THE SEISMIC RESISTANCE CATEGORY I STRUCTURES

Reference: IAEA SSR 2/1, Req. 15

The acceptance criteria shall be specified for each item important to safety corresponding to the key physical parameters including the seismic resistance to cover all operational states and accident conditions.

The acceptance criteria shall be specified in compliance with the state supervision requirements and respective national and international standards and codes.

3.3.8.2 DESIGN OF THE SEISMIC RESISTANCE CATEGORY I STRUCTURES IN THE PRELIMINARY DESIGN CONCEPT

This section contains the requirements for the design of the seismic resistance category I structures in the preliminary design concept. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant.

The reinforce concrete as well as steel structures shall be designed based on the relevant codes and standards.

The design loads shall conform to the respective norms, standards and practice and they shall be in compliance with the relevant requirements of regulatory authorities.

The structures exposed to combinations of own weight, heat load and seismic load included in the seismic category I shall be dimensioned in order to be stressed within the elastic behaviour limits.

The area of elastic behaviour shall be considered as the area limited by the yield strength of the effective load-carrying structural materials of an element.

The design of reinforce concrete elements stressed by bending shall always conform to the yield strength of the bracing steel.

Partial Preliminary Assessment

The preliminary design concept summarizing the most critical requirements for systems, structures and components specified in Section "3.3.8.2 Design of Seismic Resistance Category I Structures" creates the preconditions for meeting the requirements of IAEA SSR 2/1 [L. 252] (Req. 14, 15).

3.3.9 RESISTANCE OF MACHINERY SYSTEMS AND COMPONENTS

The section deals with the resistance of the machine technology in terms of strength, service-life and seismic calculation.

3.3.9.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE RESISTANCE OF MACHINERY SYSTEMS AND COMPONENTS

Applicable legislation, cited in the Terms of Reference, does not specify any requirements for the resistance of machinery systems and components.

This shall be specified in more detail within the next stages of the licensing documentation.

3.3.9.2 RESISTANCE OF MACHINERY SYSTEMS AND COMPONENTS IN THE PRELIMINARY DESIGN CONCEPT

The preliminary design concept assessment in the area of resistance of machinery systems and components has not been carried out because the specified range of the binding legislation does not delineate any requirements in the assessed area.

This shall be specified within the following stages of the licensing documentation.

3.3.10 SEISMIC AND DYNAMIC RESISTANCE OF THE MECHANICAL, ELECTRICAL AND I&C EQUIPMENT

The section specifies the basic requirements for the nuclear installation design in terms of the exposure to natural events, specifically with regard to the seismic and dynamic resistance of the technological equipment.

At the next stage of the licensing documentation, this section shall contain the safety proof of the seismic and dynamic design resistance at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.3.10.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE SEISMIC AND DYNAMIC RESISTANCE OF THE MECHANICAL, ELECTRICAL AND I&C EQUIPMENT

Reference: Decree No. 195/1999 Coll., Article 10, WENRA Appendix E 5.2, SÚJB Safety Guide BN-JB-1.0 (24)

- Items important to the safety of the nuclear installation shall be designed in order to provide the following in case of natural disasters that can be realistically assumed (earthquakes, windstorms, floods, etc.), or human induced events outside the nuclear installation (aircraft crash, explosions in the power plant surroundings, etc.)
  - shut the reactor down safely and maintain its sub-critical condition
  - conduct the residual reactor output away for a sufficiently long time
  - ensure that potential radioactive releases do not exceed the values stipulated by special decree (Decree No. 307/2002 Coll. [L. 4])
- In the process of nuclear installation designing the following shall be considered
  - the most serious natural events historically recorded in the respective site and its vicinity, extrapolated while taking account of the limited accuracy of values and time
  - combination of effects of natural events or human induced events and accident conditions caused by such events

3.3.10.2 SEISMIC AND DYNAMIC RESISTANCE OF THE MECHANICAL, ELECTRICAL AND I&C EQUIPMENT IN THE PRELIMINARY DESIGN CONCEPT

This section contains the characteristics of the system of provision of seismic and dynamic resistance of the mechanical, electrical and I&C equipment in the preliminary design concept. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant.

The design of mechanical, electrical and I&C equipment shall respect the basic requirements for systems, structures and components specified in Section "3.3.7.2 Seismic Resistance in the Preliminary Design Concept" and "3.3.8.2 Construction of Seismic Resistance Category I Buildings in the Preliminary Design Concept".

The scheme for seismic qualification shall be established in order to ensure that the equipment, for which the functionality or maintaining the structural integrity is required in case of the design basis earthquake, is able to perform its design function, and possibly maintain its structural integrity.

At the same time it shall be proved that in case of an earthquake of lower magnitudes than the design basis earthquake (usually half a magnitude of the design basis earthquake) the performance of the equipment is not affected negatively and that such earthquake does not result in the fatigue or ageing of the equipment that might result in the failure or worsening of the design function in case of a subsequent design basis earthquake.

Seismic and dynamic qualification of the equipment where the maintaining of functionality is required shall meet the requirements of:

- IEC 980, possibly amended with non-nuclear standards, or
- other relevant standards (i.e. standards provably equivalent to the IEC 980 standard)

The qualification shall be performed using testing, analyses, use of the experimental data for similar equipment or their combination. The qualification using analogy shall be considered acceptable for the same type of equipment. To the extent enabled by the relevant standard, the experimental data shall be used in order to particularize the results of analyses.

Seismic and dynamic qualification of the equipment where maintaining the structural integrity is required, shall be performed to the extent necessary for the respective purpose of maintaining the structural integrity. Comparison with the test results or operational experience shall be considered a sufficient qualification method.

Partial Preliminary Assessment

The preliminary design concept summarizing the most critical requirements for seismic and dynamic resistance of the mechanical, electrical and I&C equipment specified in Section "3.3.10.2 Seismic and Dynamic Resistance of the Mechanical, Electrical Equipment and I&C" creates the preconditions for meeting the requirements of the Decree No. 195/1999 Coll. [L. 266] Article 10.

3.3.11 RESISTANCE OF THE MECHANICAL, ELECTRICAL AND I&C EQUIPMENT AGAINST THE EXTERNAL ENVIRONMENT

The section specifies the requirements for the qualification of the items important to safety.

That includes the process of demonstration and documentation of the capabilities of the equipment to perform, in case of need, the required safety functions during the postulated operational conditions, including the design accidents. The equipment qualification process is intended to document and justify, in an acceptable way, the assumption that the equipment does not lose its ability to perform the function specified by the design as a result of the environmental influence to which it may be exposed.

At the next stage of the licensing documentation, the section shall contain the determining of the qualification of items important to safety at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.3.11.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE RESISTANCE OF THE MECHANICAL, ELECTRICAL AND I&C EQUIPMENT AGAINST THE EXTERNAL ENVIRONMENT

Reference: Decree No. 195/1999 Coll., Article 7, SÚJB Safety Guide BN-JB-1.0 (27)

During normal and abnormal operation, during tests and upon the arising of accident conditions, the items important to safety shall ensure that they are not damaged as a result of failures of other equipment located inside the nuclear installation. That is why they will have to be able to withstand the changes of the environment related to such failures and be properly located and adequately protected against dynamic and other effects (thrown object, line vibrations, fluid leakage, overpressure overload).

Reference: IAEA SSR 2/1 Req. 30

The qualification scheme shall be performed in order to confirm that the items important to safety are able to meet the requirements for the performance of design function in case of need and in the prevailing conditions of the external environment, throughout the whole lifetime. The conditions of the external environment arising during equipment maintenance and testing shall also be considered.

The external environmental conditions shall include the changes in the external environment which are considered in the design basis.

The qualification scheme shall include the assessment of the effects of ageing caused by various factors of the external environment (e.g. vibrations, radiation, humidity, temperature, pressure) throughout the expected equipment lifetime. If the equipment is subject to the effect of external natural events and it is required to perform the safety function during or after such event, the qualification scheme shall simulate such conditions as accurately as possible, either using a test or analysis or by combining test and analysis.

Any conditions of the external environment, which may be reasonably expected to arise during the specific operational state, e.g. during the regular testing of release from the containment, shall be included in the qualification scheme.

3.3.11.2 RESISTANCE OF THE MECHANICAL, ELECTRICAL AND I&C EQUIPMENT AGAINST THE EXTERNAL ENVIRONMENT IN THE PRELIMINARY DESIGN CONCEPT

This section contains the characteristics of the system of providing the resistance of the mechanical, electrical and I&C equipment against external effects in the preliminary design concept. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant.

The design of mechanical, electrical and I&C equipment shall respect the basic requirements for the protection against external effects specified in Section “3.3.4 Protection against External Effects in the Preliminary Design Concept”.

The equipment qualification scheme in relation to the external environment shall be established. The scheme shall ensure maintaining of the design function for the equipment where the functionality is required for the respective conditions of the external environment (normal operation, abnormal operation and design accidents).

The qualification shall be performed using the appropriate recognized standards (e.g. IEC 780 or equivalent standard for electrical equipment).

The environmental conditions shall include temperature, pressure, humidity, radiation, effects of chemicals, electromagnetic interference, ageing or flooding, but they will not be limited only to those variables. At the same time, the effect of combining of the specified environmental conditions shall be considered.

The qualification shall be performed using physical tests or using experience (i.e. the proof of similarity to the already qualified equipment or to the equipment exposed to harsher conditions).

The qualification shall be performed in the necessary extent (depending on the environmental conditions which are possible for the respective equipment) in the following or another equivalent order:

- thermal ageing
- long-term operation
- radiation ageing
- mechanical vibrations
- seismic load
- emergency exposure
- loads from a LOCA accident
- post-accident loads

In case of the equipment performing the function in the design extension conditions, the ability to perform, with a reasonable degree of certainty, the specified design functions in such conditions shall be proved.

This proof need not be performed with the same degree of conservatism as the above-specified qualification of the equipment in relation to the environmental conditions.

The proof shall consider all conditions that the respective equipment is exposed to in the design extension conditions (as a result of the initiating event proper or environmental conditions, e.g. temperature, pressure, radiation, etc.).

The scheme of maintenance, inspections and testing and necessary replacement of the qualified equipment shall be set in order to ensure maintaining of the required ability to perform the design function (or maintain the structural integrity) throughout the whole power plant lifetime.

Partial Preliminary Assessment

The preliminary design concept summarizing the most critical requirements for the resistance of mechanical, electrical and I&C equipment against the external environment specified in Section "3.3.11.2 Resistance of the Mechanical, Electrical Equipment and I&C against the External Environment" creates preconditions for meeting the requirements of the Decree No. 195/1999 Coll. [L. 266] Article 7 and document IAEA SSR 2/1 [L. 252] Req. 30.

3.3.12 PIPING ROUTE DESIGN ASSESSMENT

The section deals with the strength, service-life and seismic calculations of pipeline systems.

3.3.12.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE PIPING ROUTE DESIGN ASSESSMENT

This shall be specified in more detail within the next stages of the licensing documentation.

3.3.12.2 PIPING ROUTE DESIGN ASSESSMENT IN THE PRELIMINARY DESIGN CONCEPT

The preliminary assessment of the design concept in the area of the piping route assessment has not been carried out because the specified range of the binding legislation does not delineate any requirements in the assessed area. This shall be specified within the following stages of the licensing documentation.

3.3.13 BOLTS (THREADED CONNECTIONS)

The section specifies the basic requirements for the materials used in the reactor coolant system.

Bolts are used as connecting components for joining and sealing of dividing surfaces of main components of the primary coolant circuit – reactor, steam generator, pressurizer and other flange joints in the primary and secondary coolant circuit.

At the next stage of the licensing documentation, this section shall contain the selection of materials, structures, manufacture, testing and inspection of bolted joints of the selected equipment throughout the whole equipment lifetime.

3.3.13.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE DESIGN OF BOLTS (THREADED CONNECTIONS)

Reference: Decree No. 195/1999 Coll., Article 22 (2)

The design of the primary coolant circuit specifies the materials verified for those purposes and in compliance with the respective codes, technical standards or technical conditions and it shall document their sufficient dimensioning using theoretical calculation and experimental verification.

The design shall include a margin for degradation of the properties of the material that may take place during the operation due to erosion, corrosion, material fatigue, chemical environment, irradiation and ageing, speed of degradation of their properties and it shall also define the method of proving the quality of production using the modern methods available and it shall specify the scheme and methods of examining their state during operation.

3.3.13.2 BOLTS (THREADED CONNECTIONS) IN THE PRELIMINARY DESIGN CONCEPT

This section contains the requirements for bolts (threaded connections) in the preliminary design concept. The information for the specification of the design characteristics was based on the technical part of the tender documentation stipulating the requirements for safety and technical design of the future power plant.

Bolts (threaded connections) shall meet the basic requirements specified in Section "3.3.9.2 Resistance of Machinery Systems and Components in the Preliminary Design Concept".

Bolted connections providing the sealing of pressure boundary in the places of occurrence of contaminated medium shall be designed in order to minimize the possibility of a release of the medium through the connections and provide the detection and monitoring of the release of the medium.

In order to minimize the effects of such release, the materials resisting to corrosion shall be used for bolted connections. These materials shall not threaten the functionality of the bolted connection.

The bolted connection on the primary coolant circuit equipment shall be documented by calculation which also specifies the tightening torque.

Partial Preliminary Assessment

The preliminary design concept summarizing the most important requirements for bolts (threaded connections) specified in Section "3.3.13.2 Bolts (Threaded Connections)" creates the preconditions for meeting the requirements laid down in the Decree No. 195/1999 Coll. [L. 266] Article 22 (2).

3.3.14 SUMMARY PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT IN THE AREA OF THE DESIGN OF STRUCTURES, COMPONENTS, SYSTEMS AND EQUIPMENT

The design of ETE3,4 shall meet the safety requirements specified in Section "3.3.1 Compliance with Basic Requirements of State Supervision in Order to Provide Safety" and at the same time the requirements specified in Sections 3.3.(2-13).1, which are based on the requirements stipulated by the Act No. 18/1997 Coll. [L. 2] and its implementing decrees for nuclear safety, radiation protection and emergency preparedness as well as on the requirements specified in IAEA SSR 2/1 [L. 252], requirements of WENRA [L. 27] and [L. 270] and SÚJB Safety Guide BN-JB-1.0 [L. 276].

To meet those requirements, the ETE3,4 shall use systems, structures and components and their equipment, which shall ensure (with adequate reliability and resistance) the technological functions relevant in the qualitative and quantitative way in accordance with the specified requirements for the prescribed safety functions.

The principles of the design solution characterized in Sections 3.3.(2-13).2 specifying the basic requirements to provide safety for the purposes of the preliminary design concept were created based on the requirements submitted by the licence applicant applied to potential suppliers of the nuclear installation within the tender and they make up the concept of the design solution for this part of the design. The partial assessments performed prove that the expected design creates preconditions for compliance with the specific requirements for systems, structures and components as well as the safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], SÚJB Safety Guide BN-JB-1.0 [L. 276], documents IAEA SSR 2/1 [L. 252] WENRA [L. 27] and WENRA NEW [L. 270].

The systems, structures and components and their equipment comply with the performed preliminary assessment of the design concept and create the preconditions to fulfil the defined scope of basic safety rules and principles in the design of the future new nuclear installation in the Temelín location.

The particular method of implementation of the individual safety requirements specified in Sections 3.3.(2-13)X.2, shall be specified in detail in the nuclear power plant design.

3.4 REACTOR

This comprehensive part of the Initial Safety Analysis Report is structured into three basic sections:

The introductory Section 3.4.1 summarises, analyses and specifies basic legislative requirements for the reactor equipment, including description of the fuel system, its nuclear, thermal, and hydraulic parameters, requirements for the reactor materials, and description of the reactivity control systems.

Section 3.4.2 that follows contains description and specification of the basic requirements for the performance of the reactor equipment specified in the introductory Section 3.4.1, in a form that summarises the applied design requirements for the relevant reactor equipment subsystems in relation to all the relevant designs involved in the tender. The objective of the section is to formulate the general characteristics of the project for the purposes of the partial preliminary procedure.

The final Section 3.4.3 contains a comprehensive preliminary evaluation of the reactor equipment design concept, summarising the conclusions of the partial preliminary evaluations completed in Section 3.4.2. Within the assessment of the summary of requirements so obtained, the section contains a preliminary evaluation of the design concept as required by the law. At the next stage of the licensing documentation the applicant shall provide evaluating and supporting information, allowing assessment of the reactor equipment's ability to perform its safety functions throughout the entire reactor service life in all states considered in the design requirements, including steady-state and transient operation modes, also under the design extension conditions.

3.4.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE REACTOR

3.4.1.1 GENERAL REQUIREMENTS FOR THE PRIMARY CIRCUIT

References: Decree No. 195/1999 Coll., Article 22(1)

The primary circuit and its auxiliary, control and protective systems shall be designed so that:

- the required strength, service life, and reliability of performance of their parts and equipment are ensured with an adequate margin in both normal and abnormal operating conditions
- no inadmissible coolant leaks take place
- they are adequately resistant against the development of failures and make for a slow evolution and timely detection of any failures
- large-scale failures are ruled out
- primary circuit components that contain the coolant, such as the reactor pressure vessel, pressure piping, pipes and their coupling, valves and sealing and washers, including their fastening, withstand static as well as dynamic stresses expected in any operating state and in accident conditions

References: SÚJB Safety Guide BN-JB-1.0 (67)

The pressure and coolant circuit shall be designed, manufactured and tested at a quality adequate to its safety function.

3.4.1.2 REQUIREMENTS FOR THE FUEL SYSTEM

This subsection contains the basic as well as specific design principles and requirements for the fuel system stipulated by selected legislation. At the next stage of the licensing documentation, this section shall include those design principles applied to the fuel system's strength, thermal, and chemical design that may have an impact on the safety and reliability of the plant operation, as well as the methods of their implementation. Among other things, this section shall include the design strength limits, an overview of the properties of the materials, associated assessment of the design for the various components, etc.

General design requirements

References: Decree No. 195/1999 Coll., Article 3; SÚJB Safety Guide BN-JB-1.0 (56)

The reactor core and associated coolant, control and protective safety systems shall ensure (with an adequate margin) that the specified acceptance criteria for the reactor core are not exceeded in any state considered by the design.

References: Decree No. 195/1999 Coll., Article 13(3); SÚJB Safety Guide BN-JB-1.0 (52)

The mechanical parts constituting the reactor core shall be designed so that they are able to withstand static as well as dynamic loads and do not preclude safe reactor shutdown and adequate reactor core coolant in normal or abnormal operation states as well as during any event postulated by the design, except for adequate heat removal during severe accidents accounted for in the design.

Specific design requirements

References: Decree No. 195/1999 Coll., Article 14(1); SÚJB Safety Guide BN-JB-1.0 (52); IAEA SSR 2/1 2.14, Req. 43, 6.3

The fuel system and any mechanical parts in its vicinity, including their fastening, shall be designed so that they are able, under postulated design conditions, to withstand wear and maintain their structural and dimensional integrity and withstand changes in physical properties and expected radiation effects and effects of the environmental conditions on the materials properties.

References: Decree No. 195/1999 Coll., Article 13(2); SÚJB Safety Guide BN-JB-1.0 (55); IAEA SSR 2/1, Req. 43, 6.1

The following effects in relation to the processes of the material properties degradation and the deterioration of the environmental conditions shall be considered: differences in the dilatation and deformation of the materials; external coolant pressure; internal pressure increase inside the fuel elements due to fission products release; irradiation of the fuel and the other fuel assembly materials; pressure and temperature changes due to power variations; chemical effects; static and dynamic stresses,, including stress from the coolant flow and from mechanical oscillations; changes associated with heat transfer, which can occur due to deformations and/or chemical effects.

Adequate margins shall be applied in all evaluations to account for data and calculation uncertainties and manufacturing tolerances.

References: Decree No. 195/1999 Coll., Article 14(3); SÚJB Safety Guide BN-JB-1.0 (54); WENRA App. E 7.2; IAEA SSR 2/1, Req. 43, 6.2

Acceptance criteria for the fuel shall be set for both normal and abnormal operation in accordance with the general design requirements. The design of the fuel system shall guarantee that the criteria are not exceeded under relevant design conditions. Moreover, conditions that may occur in the reactor core during anticipated operational occurrences must not cause significant deterioration of the fuel system's design parameters, so that the fuel system remains suitable for subsequent operation. Any fission product leak shall be kept under the minimum level which is reasonably achievable.

References: Decree No. 195/1999 Coll., Article 14(4); SÚJB Safety Guide BN-JB-1.0 (52); IAEA SSR 2/1, Req. 44

Under accident conditions resulting from any initiating event which is postulated in the design, the fuel system and any mechanical parts in the vicinity of the reactor core (including their fastening and supporting structures) shall maintain their desired geometry and shall not suffer such damage that would prevent insertion of the control rods or preclude actuation of any other systems for reactivity control and reactor shutdown or would preclude efficient heat removal from the reactor core except for adequate heat removal during severe accidents accounted for in the design.

References: Decree No. 195/1999 Coll., Article 14(5); SÚJB Safety Guide BN-JB-1.0 (54); WENRA App. E 7.2

Acceptance criteria for the fuel shall be set for each event postulated in the design in accordance with the general design requirements. It shall be demonstrated that the design of the fuel system ensures that the criteria will not be exceeded.

References: Decree No. 195/1999 Coll., Article 14(6); SÚJB Safety Guide BN-JB-1.0 (53)

The fuel assembly design shall enable parts of the assemblies to be adequately inspected.

References: Decree No. 195/1999 Coll., Article 14(7); SÚJB Safety Guide BN-JB-1.0 (53)

The chosen design solution and the design of the fuel assembly components shall be adequately documented by experiments or by operation experience.

3.4.1.3 REQUIREMENTS FOR THE NUCLEAR PARAMETERS

This part of the Initial Safety Analysis Report describes both basic and specific requirements for the nuclear design of the fuel system as set by applicable legislation. In the next step of the licensing documentation, this section shall describe the design principles of the nuclear design and reactivity control systems, including limiting values and ways to comply with them. The assessment shall contain analysis of parameters such as operational reactivity margin, fuel burnup, design cycle lengths and related parameters of the fuel reloads, efficiency of burnable poisons, reactivity feedbacks and reactivity coefficients, power distribution control, power distribution stability, control of reactivity insertion by the mechanical and chemical systems, location and reactivity weight of individual groups of control rods, subcriticality margin after shutdown and efficiency of the safety systems, efficiency of a stuck control rod, subcriticality during fuel storage and handling, pressure vessel irradiation, etc.

General design requirements

References: Decree No. 195/1999 Coll., Article 13(1); SÚJB Safety Guide BN-JB-1.0 (56)

The reactor core and associated coolant, control and protection safety systems shall ensure (with an adequate margin) that the specified acceptance criteria for the reactor core are not exceeded in any state considered by the design.

References: Decree No. 195/1999 Coll., Article 13(2); SÚJB Safety Guide BN-JB-1.0 (57)

The reactor core and the associated coolant, control and protection safety systems shall be designed so that the resulting effect of the instantaneous reactivity feedbacks in the reactor core should counteract a rapid reactivity increase in any operating state with a critical reactor.

Specific design requirements

References: Decree No. 195/1999 Coll., Article 15(2); SÚJB Safety Guide BN-JB-1.0 (61); IAEA SSR 2/1, Req. 45, 6.4.

The selected design shall be equipped with means for the detection of the level and distribution of the neutron flux; those means shall be capable of detecting regions of the reactor core where the neutron flux level and distribution might cause the reactor core acceptance criteria to be exceeded. The design of the reactor core and the nuclear design of the fuel system, along with the control system, shall enable the neutron flux level and distribution to be kept within specified limits in any reactor core state during normal or abnormal operation.

References: Decree No. 195/1999 Coll., Article 15(1); SÚJB Safety Guide BN-JB-1.0 (56); IAEA SSR 2/1, Req. 45

The design requirements for the fuel system specified in Section 3.4.1.2 shall be maintained for any neutron flux level and distribution that might occur in any considered design states of the reactor core, including states following shutdown, states during or after fuel reloading, and states occurred during anticipated operational occurrences or in accident conditions arising from any initiating event considered by the design (except for severe accidents). The adequate requirement set for the considered design state applies.

References: IAEA SSR 2/1, Req. 45

The neutron flux distributions that may occur in the above-mentioned design states shall be inherently stable and, at the same time, the requirements for control system actions needed to maintain the neutron flux level, distribution and stability within the specified limits in any operation states shall be minimized.

References: Decree No. 195/1999 Coll., Article 15(3); SÚJB Safety Guide BN-JB-1.0 (58)

The reactor core and the associated coolant, control and protection systems shall be designed so as to ensure that power oscillations which might cause the specified design fuel limits to be exceeded are excluded or are reliably and immediately detected and suppressed.

References: Decree No. 195/1999 Coll., Article 21(1); SÚJB Safety Guide BN-JB-1.0 (62); IAEA SSR 2/1, Req. 46, 6.7

The selected design shall include reactivity and reactor shutdown systems which are capable of shutting down the reactor during normal and abnormal operation and under accident conditions arising from any event considered by the design. The systems shall maintain the reactor in the shutdown state even in a situation causing the highest reactor core reactivity. The efficiency, promptness of action and shutdown margin shall ensure that the specified acceptance criteria are not exceeded.

References: Decree No. 195/1999 Coll., Article 21(2); SÚJB Safety Guide BN-JB-1.0 (62); WENRA App. E 9.5; IAEA SSR 2/1, Req. 46, 6.9.

The means for reactivity control and reactor shutdown shall comprise at least two independent systems based on different principles and capable of performing their functions even in the case of a single failure.

References: Decree No. 195/1999 Coll., Article 21(3); SÚJB Safety Guide BN-JB-1.0 (63); WENRA App. E 9.6

At least one of those systems shall meet the requirements for the safety system, specified in Section 3.3.1. This system itself shall be capable of getting the reactor rapidly from a normal or abnormal state or accident conditions arising from any event considered by the design into the subcritical state with an adequate margin, even if assuming a single failure.

References: Decree No. 195/1999 Coll., Article 21(4); SÚJB Safety Guide BN-JB-1.0 (63); IAEA SSR 2/1, Req. 46, 6.10.

At least one of those systems shall be able itself to bring the reactor from the normal operating state to a subcritical state and to maintain it in the subcritical state with a sufficient margin, even in a situation causing the highest reactor core reactivity.

References: Decree No. 195/1999 Coll., Article 21(8); SÚJB Safety Guide BN-JB-1.0 (66)

A part of systems dedicated for reactor shutdown will also be used for reactivity control or for neutron flux shaping while constantly maintaining a sufficient margin for putting the reactor to a subcritical state.

References: SÚJB Safety Guide BN-JB-1.0 (60), IAEA SSR 2/1 Req. 45, 6.6.

The selected design shall ensure that under both normal and abnormal operation conditions as well as in accident conditions due to any of the events considered in the design, the total positive reactivity insertion and the rate of its change are limited or compensated for to avoid any subsequent damage to the pressure boundary of the reactor coolant system, maintain the coolant ability, and avoid severe damage to the fuel system except for severe accidents considered in the design. Design basis events associated with reactivity change shall include control rod ejection and/or drop, steam generator steam pipe break, coolant temperature and pressure change, and cold water ingress.

References: Decree No. 195/1999 Coll., Article 21(5); SÚJB Safety Guide BN-JB-1.0 (64); WENRA App. E 8.4; IAEA SSR 2/1, Req. 46, 6.8

When demonstrating the required characteristics of systems for reactor shutdown, special consideration shall be given to failures arising anywhere at the nuclear installation that could render part of the means of shutdown out of operation (e.g. failure of control rod to insert) or that could result in a common-cause failure.

References: WENRA App. E 8.4

As an additional conservatism, one stuck control rod shall be considered in the analysis of the design events. The objective of this assumption is to ensure that the requirement for demonstration of an adequate margin to criticality following shutdown is met. If a stuck control rod is found to be the most severe single failure, then the design events analyses may postulate a stuck control rod as a separate single failure. The analyses shall consider a stuck control rod with the highest reactive efficiency in the hot state conditions with a zero power. Furthermore, a conservative process of insertion of reactivity following initiation of the reactor trip chain shall be assumed, including conservative time delays.

References: Decree No. 195/1999 Coll., Article 21(6); SÚJB Safety Guide BN-JB-1.0 (59); IAEA SSR 2/1, Req. 46, 6.11, Req. 45, 6.5

The means for reactivity control and reactor shutdown shall be adequately designed to be able to prevent any foreseeable and undesirable criticality increase and to avoid or minimise the impact of a spontaneous occurrence of a critical condition. This requirement shall be met for any routine or unusual activities considered by the design, where the reactor core reactivity can increase while the reactor is shut down (e.g. when removing a control cluster for maintenance or for fuel reloading) or following postulated initiating events, and the acceptance criteria for the fuel system shall not be exceeded. These requirements shall be met even considering a single failure of those systems.

The design of the means for reactivity control and reactor shutdown shall take into account issues of mechanical and other wear and irradiation-induced changes, such as burnup, gas release, and changes in the physical properties of the nuclear fuel and of the materials of fuel system components in the reactor core.

References: Decree No. 195/1999 Coll., Article 21(7); SÚJB Safety Guide BN-JB-1.0 (65); IAEA SSR 2/1, Req. 46, 6.12

The means for reactivity control and reactor shutdown shall be maintained in a condition ensuring that they will perform their safety function in any operation state and in accident conditions. For this purpose the selected design solution shall be equipped with appropriate instrumentation, and procedures shall be specified for testing the performance of those systems.

3.4.1.4 THERMAL AND HYDRAULIC PROPERTIES

The thermal and hydraulic design of the reactor core assures adequate heat transfer, compatible with the heat source distribution within the reactor core. This is ensured by adequate heat removal by the reactor coolant system or by the emergency core coolant system if used.

The reactor systems shall meet the requirements of legislation applicable to the reactor core and technological equipment involved in heat removal, specified later in the text. In the next step of the licensing documentation, this section shall contain a description of the concept of the core's thermal and hydraulic parameters, including safety assessment with respect to the occurrence of heat transfer crisis and specification of the linear thermal power, determination of the stream fraction in the reactor core, determination of the coolant flow distribution inside the core, and determination of the pressure losses and hydraulic loads. Furthermore, this section shall contain a description of the thermal and hydraulic parameters of the reactor coolant system design and coolant system configuration, including operation diagrams.

References: Decree No. 195/1999 Coll., Article 8(1)

The technological assemblies and equipment involved in the removal of heat released by nuclear fission, residual heat, and operational heat, shall reliably ensure adequate reactor cooling in both normal and abnormal operating conditions as well as in accident situations.

References: Decree No. 195/1999 Coll., Article 13(1); SÚJB Safety Guide BN-JB-1.0 (56)

The reactor core and the associated coolant, control and protection systems shall assure with a sufficient margin that the specified design limits will not be exceeded in any state accounted for by the design.

References: Decree No. 195/1999 Coll., Article 13(2); SÚJB Safety Guide BN-JB-1.0 (57)

The reactor core and the associated coolant, control and protective systems shall be designed so that the resulting effect of the instantaneous reactivity feedback in the reactor core should counteract a rapid reactivity increase in any operating state with a critical reactor.

3.4.1.5 REQUIREMENTS FOR THE MATERIALS OF THE REACTOR AND OF THE PRIMARY COOLANT CIRCUIT SYSTEM

This section specified general requirements for materials to be used within the design of the structures, systems, and components of the plant's primary coolant circuit. In the next step of the licensing documentation, this section shall include requirements for the materials of the control cluster drive systems and for the materials of the reactor internals to a depth of detail and in the structure required by RG 1.206 [L. 275].

None of the documents referred to and/or analysed specifies particular conditions or requirements put directly on the materials of the reactor.

References: Decree No. 195/1999 Coll., Article 22(2a),(2c); IAEA SSR 2/1, req. 47

The primary coolant circuit system design shall:

- specify such materials as have been proven for these purposes and comply with applicable codes, technical standards and/or technical specifications
- include a margin accounting for degradation of the properties that may take place during operation due to erosion, corrosion, material fatigue, chemical environment/medium, irradiation, and ageing, and a margin to account for the uncertainty of the identification of the initial condition of the components and rate of degradation of their properties

References: SÚJB Safety Guide BN-JB-1.0 (52)

The fuel system and/or mechanical parts located near the reactor core, including their fastening, shall be designed so as to withstand both static and dynamic effects and wear, and to keep their structural and dimensional integrity and withstand changes in the physical properties as well as expected radiation effects and effects of the surrounding environment/medium on the properties of the materials in both normal and abnormal operating conditions as well as during any initiating event postulated in the design, so that they should not stand in the way of a safe reactor shutdown or reactor core cooling.

References: SÚJB Safety Guide BN-JB-1.0 (55)

The material property degradation processes and conditions to be considered shall include the effect of the external coolant pressure; increased internal pressure inside the fuel elements due to fission products; irradiation of the fuel and the other fuel assembly materials; pressure and temperature changes due to power variations; chemical effects; static and dynamic stresses, including stresses from coolant flow and from mechanical oscillations; and changes in heat transfer which can occur due to deformations and/or chemical effects. Adequate margins shall be applied in all evaluations to account for data and calculation uncertainties and manufacturing tolerances.

References: SÚJB Safety Guide BN-JB-1.0 (70)

The design shall include means for a prompt detection of any coolant leak and enable periodic inspections and tests of the reactor's pressure and coolant circuits, including assessment of the reactor pressure vessel material.

3.4.1.6 PROPERTIES OF THE REACTIVITY CONTROL SYSTEMS

Nuclear power plants use, in both normal and abnormal operating conditions, two independent reactivity control systems based on two different principles, either being able, independently of the other, to ensure transition of the reactor system from any normal operating state to a subcritical state and to maintain it subcritical at the coolant's working temperature. Reactivity change also controls reactor output.

Two methods exist to change reactivity during normal or abnormal operation:

- By displacing the absorber by means of mechanical components of the reactor control and protection system
- By changing the absorption properties of the moderator, achieved through the primary coolant makeup and purification system

In accident conditions, reactivity change is also contributed to by the emergency core coolant system and the similar systems for boric acid addition to the primary circuit in accident situations.

The reactivity control system shall meet the legislative requirements specified below. In the next step of the licensing documentation, this section shall contain a functional design of the reactor control systems to a depth of detail and in the structure set out in RG 1.206 [L. 275].

References: Decree No. 195/1999 Coll., Article 21; IAEA SSR 2/1 Req. 45, 6.5, Req. 46, WENRA App. E 9.5, 9.6; SÚJB Safety Guide BN-JB-1.0 (60, 62, 63, 64, 65, 66)

- The reactor shall be equipped with systems capable of shutting it down during both normal and abnormal operating conditions as well in accident situations. The systems shall keep the reactor shut down even in situations bringing about the highest reactor core reactivity. The shutdown efficiency, promptness and margin shall be such that the specified design limits will not be exceeded.

- The reactor shutdown system shall consist of two or more independent systems based on different principles and capable of performing their functions even in the event of a single failure.
- At least one of the systems referred to in paragraph 2 shall on its own be able to transfer the reactor promptly from the normal or abnormal operating state or accident state to a subcritical state with an adequate margin in a situation assuming a single failure.
- At least one of the systems referred to in paragraph 2 shall on its own be able to transfer the reactor from normal operation to a subcritical state and to maintain it in the subcritical state with an adequate margin in a situation bringing about the highest reactor core reactivity.
- When demonstrating the required properties of the system as regards reactor shutdown, special attention shall be paid to failures arising anywhere within the nuclear installation and capable of putting a part of the facility out of operation.
- The reactor shutdown systems shall be able to prevent spontaneous occurrence of a critical condition. This requirement shall also be met during activities that increase reactivity while the reactor is shut down (e.g. when a control cluster has been removed for maintenance or fuel reloading), even in a situation of a single failure of the systems.
- The measuring systems and tests shall ensure that the reactor shutdown systems are in the required condition.
- When the reactor is operational, a part of the reactor shutdown systems may be used for reactivity control or for neutron field shaping, provided that the margin for shutdown is maintained constantly.

SÚJB Safety Guide BN-JB-1.0 (60)

Design basis events associated with reactivity change shall include control cluster ejection and/or drop, steam generator steam pipe break, coolant temperature and pressure change, and cold water ingress.

3.4.2 DESCRIPTION AND PROPERTIES OF THE REACTOR SYSTEM DESIGN FOR THE PURPOSE OF THE PRELIMINARY ASSESSMENT

This section describes the design features for the preliminary assessment of the design concept. The properties of the design were identified based on the technical part of the reference documentation setting out requirements for the safety and technological aspects of the design of the future power plant. This section includes partial assessment of whether the preliminary concept of the design segment in question complies with the requirements specified in Sections 3.4.1.1 through 3.4.1.6. The objective is to assess whether the general level of the requirements for the functions of the reactor systems is met. The specific method of technical implementation of the systems will only be specified in the NPP design. Subsequently, at the next stage of the safety documentation, evidence of the safety relevance of the design solution shall be submitted.

The reactor system shall be designed so as to ensure satisfaction of the basic safety requirements and principles specified in Section 3.3.1. These include, in particular, the principle of defence in depth and the acceptance criteria specified for the systems, structures and components specified within the design basis, including the relevant design requirements based on the conditions of the site as specified in Chapter 2 and summarised for the most important ones in Section 2.10.

The design of the reactor shall meet the requirements for the safety functions stipulated by applicable legislation.

The design solution selected for implementation need not include all systems, structures and components specified in this section. However, if the resulting design uses any of the systems, this shall comply with the relevant requirements specified in this section.

3.4.2.1 A COMPREHENSIVE DESCRIPTION OF THE REACTOR SYSTEM IN THE PRELIMINARY CONCEPT OF THE DESIGN

This section provides a comprehensive overall description of the reactor system in the preliminary concept of the design. The description was developed based on the technical part of the tender documentation stipulating the requirements for safety and technical design of the future power plant.

The reactor facility is a complex of nuclear power plant systems and elements designed to convert nuclear energy into heat. It includes the reactor, reactor internals and directly related systems required for its normal operation, abnormal operation, emergency cooling, emergency protection and keeping in a safe condition, provided that the necessary supporting functions are provided by other plant systems. The boundaries of the reactor facility are defined for each power plant by the general designer and general supplier.

The reactor facility shall comply with regulatory requirements for the primary coolant circuit systems and their auxiliary, control and safety systems specified in the sections below. In the next stage of the licensing documentation, this section shall contain a specific description of the mechanical strength, nuclear, thermal, and hydraulic parameters of the design for each reactor component, including fuel system, reactor core, reactor internals, and the reactivity control system, with respect to the design solution chosen. The description shall include both the independent and the mutually affected performance and safety functions of each component.

The requirements for the reactor facility materials, which are based on the agreed-on legislative sources for the development of the Initial Safety Analysis Report, are described in detail in Section 3.4.2.5.

Additional components of the reactor facility shall provide for reactor control and protection in the process of heat transfer to the coolant inside the reactor core and its subsequent transport to the steam generator. The requirements for the reactor facility materials, which are based on the agreed-upon legislative sources for the development of the Initial Safety Analysis Report, are described in detail in Section 3.4.2.5.

Reactor

The reactor is the source of heat in the nuclear power plant. It is designed for heating the coolant by the energy generated by the controlled chain fission reaction of the nuclear fuel in the reactor core.

This includes the reactor pressure vessel, consisting of the vessel itself and its head, reactor internals located inside the reactor pressure vessel (core barrel, neutron reflector, ...), and control rod drives and instrumentation installed on the reactor head. The requirements for the reactivity control systems, which are based on the agreed-upon legislative sources for the development of the Initial Safety Analysis Report, are described in detail in Section 3.4.2.6.

The main functions of the reactor include the mounting of the core, providing an adequate amount of moderator (also serving as the coolant in PWRs) required to maintain the chain fission reaction in the core, and maintaining the primary circuit tightness.

The coolant enters the reactor through the inlet nozzles, flows through the circular gap between the reactor pressure vessel body and the core barrel, and enters the core from below. While passing through the reactor core, the coolant absorbs heat released by the nuclear fuel fission reaction. The requirements for the thermal and hydraulic design, which are based on the agreed-upon legislative sources for the development of the Initial Safety Analysis Report, are described in detail in Section 3.4.2.4.

Fuel system and its nuclear properties

The fuel system consists of a set of fuel assemblies and other reactor core components. The design and functions of the fuel system components, as described in detail in Section 3.4.2.2, shall comply with all requirements stipulated by selected legislation. The fuel system shall be designed so as to meet the requirements for mechanical strength, thermal and chemical stability, mechanical resistance and reliability of performance in accordance with the acceptance criteria based on the design principles specified in Section 3.4.2.2. The fuel system shall be compatible with the reactor internals and other systems of the reactor facility and its operation conditions, including accident conditions. The fuel system shall enable the processes occurring in the reactor facility to be monitored and controlled and the fuel system components to be handled, including storage. The materials and design solutions for the fuel system components shall be based on proven and analytically sound solutions.

The nuclear design of the fuel system, as described in detail in Section 3.4.2.3, shall create basic conditions for adherence to the design principles of the specific reactor core. At the same time, however, it shall create conditions for the required flexibility of the reactor facility, e.g. variable fuel cycle lengths, different core design strategies, operation in the variable plant load regime, etc. The properties of the specific nuclear fuel design shall be determined by the required neutron physics properties of the reactor core to be designed. The acceptance criteria at the level of the core neutron physics parameters shall be based on generally recognised, proven, and analytically sound approaches. The nuclear design shall ensure the required reactivity reserve, stability and weight of the reactivity feedbacks, efficiency and promptness of reactivity insertion by the mechanical and chemical systems, power distribution uniformity and stability in the reactor core, subcriticality during fuel handling and storage, and many others, including derived criteria specified for the selected fuel system design solution

3.4.2.2 THE FUEL SYSTEM IN THE PRELIMINARY CONCEPT OF THE DESIGN

This section describes and briefly assesses the properties of the fuel system in the preliminary concept of the design. The properties of the design were identified based on the technical part of the bid documentation stipulating the requirements for safety and technical design of the future power plant. The assessment uses principles and approaches which are routinely applied to similar projects.

This section is divided into parts dealing with the concept of the fuel assembly and the concepts of the reactivity control assemblies and other reactor core components.

The term "fuel system" is used to denote fuel assemblies, in which the controlled fission chain reaction proceeds, and other reactor core components needed to control reactivity and ensure the design characteristics of the core. A fuel assembly consists of a bundle of fuel elements (rods) arranged and fixed by the assembly structural components (such as spacer grids, guide tubes, end nozzles) constituting the fuel assembly skeleton. Reactor core components also include assemblies of absorber rods of the mechanical reactivity control system, assemblies of absorber rods of discrete burnable absorbers, assemblies of neutron source rods and sets of hydraulic plugs of the guide tubes.

Design and function of a fuel assembly

A fuel assembly consists of a defined number of fuel elements distributed across the assembly at a fixed spacing of the grid along with the guide tubes. A fuel element consists of a so called cladding tube, hermetically closed with end plugs and accommodating pellets of the nuclear material forming a stack inside the cladding tube. A fission chain reaction occurs in the nuclear material. The heat released is transferred through the gap between the pellets and cladding and subsequently through the cladding wall into the coolant. The fuel element and guide tube spacing is maintained by the spacer grids. The spacer grids are fastened to the guide tubes in several axial positions. The spacer grids form cells through which the fuel elements pass vertically. The spacer grid design also contributes to coolant mixing between the cells. Both ends of the guide tubes are inserted into fuel assembly top and bottom nozzles allowing the fuel assembly to be axially fixed and correctly oriented within the reactor internals. The guide tubes serve for reactivity control rods insertion, discrete burnable absorber rods insertion, insertion of neutron source rods and the guide tubes' hydraulic plugs. Selected guide tubes can also serve to insert in-reactor instrumentation.

Design and function of the reactivity control assembly

The reactivity control assembly (usually called control rod or control cluster) provides introduction of negative or positive reactivity by a mechanical means, i.e. axial motion within the reactor core. This assembly contains an upper head (usually called spider) to which the rods containing the neutron absorber are attached. The upper head allows the assembly to be gripped to a specific drive and to be spatially adjusted within the fuel assembly head (when inserted completely) and within the reactor internals above the core (while being withdrawn). The rods with the absorber move inside the fuel assembly guide tubes. Apart from the total reactivity control, selected assemblies also control the power distribution shape, or local neutron flux density.

Design of other reactor core components

The design of the assemblies of discrete burnable absorbers, neutron sources and hydraulic guide tube plugs is similar to that of the reactivity control assemblies. Particulars such as the number of rods, material composition, geometry, and fastening to the upper head are specific to the individual fuel system. If those components are used in the fuel system, they remain in their bottom positions for the whole time for which the fuel remains inside the core. They can be moved within the core or withdrawn from it completely during refuelling. Discrete burnable absorbers absorb neutrons and serve to compensate for excess reactivity especially at the beginning of cycle after core start-up. Neutron sources release thermal neutrons, thus increasing the indicated neutron flux level in zero power states. Hydraulic plugs limit coolant flow through unoccupied guide tubes, thus reducing parasitic fuel element bypass coolant flow.

Fuel system

The fuel system shall be designed so as to meet the basic safety requirements and principles described in Section 3.3.1. In particular, the principle of defence in depth and the acceptance criteria set for the fuel system and for the design requirements must be complied with.

The basic safety principles are detailed into the design requirements. Subsequently, acceptance criteria shall be defined at the level of each component and functional interrelations among the components. The acceptance criteria and demonstration of their satisfaction shall be documented for the selected fuel system at the next level of the safety assessment of the design.

At the present stage of assessment of the preliminary concept of the design, we shall reduce our assessment to the key areas of the design requirements for each important fuel system component.

Assessment of the preliminary concept of the fuel assembly

The proposed design of the fuel elements, fuel assemblies and reactor core components satisfies the basic safety functions, i.e. bringing the reactor to the subcritical state, adequate heat removal from the reactor core and limitation of the radiological consequences in all design basis states, including design basis accidents and design extension conditions, except for adequate heat removal during severe accidents considered in the design. In other words, the system must prevent any fuel element damage/fuel assembly deformation which might preclude insertion of the mechanical control clusters into the reactor core, reduce coolant flow to below a predetermined limit, and/or disturb the design function of other components inside or outside of the reactor.

Further, the fuel elements, fuel assemblies and reactor core components are designed in such a way that they should not undergo damage under normal or abnormal operation conditions, in other words, so that the relevant acceptance criteria for damage shall not be exceeded.

The importance and role of the reactivity control assemblies and other reactor core components in ensuring compliance with the above design requirements are detailed in another part of this section devoted to the assessment of the preliminary concept of the reactivity control assemblies and of other reactor core components.

The fact the fuel elements will not be damaged under the defined conditions is ensured by the selected fuel system / reactor core design solution, which precludes exceeding the limiting values of the design parameters, such as:

- fuel temperature
- internal pressure of gases released by the fission process
- cladding material stress, distortion and fatigue
- oxidation, hydridation, and others

The fact the fuel assembly will not be damaged under the defined conditions is ensured by the design solution, which precludes exceeding the limiting levels of the design parameters, such as:

- strain, stress and distortion of the structural elements at various design basis loads
- failure of the structural connections, etc.
- and prevents non-permitted interaction with the reactor internals, reactor core components and the storage and handling system.

For this purpose, the fuel element/assembly components are appropriately designed. It concerns, for instance, the dimensions and material properties of the fuel pellets, pellet-cladding gap, inert gas pressure inside the cladding tube, type and arrangement of the spacer grids, connections between the spacer grids and guide tubes, guide tube dimensions and material, type and properties of suspension in the fuel assembly head, etc.

The fuel elements and fuel assemblies are designed in such a way that their acceptance criteria are met for a degree of burnup which is equal to the maximum design burnup.

The fuel rod design accounts for all relevant effects such as changes in material density and dimensions, gaseous fission product release, cladding creep, and other physical properties being changed with the burnup.

The design, geometry and fastening of the guide tubes within the fuel assembly skeleton enables the absorbing elements of the control clusters to be safety inserted. The design of the fuel assembly skeleton and the arrangement of the fuel elements in the spacer grids assure adequate coolant flow and mixing inside the reactor core.

The design of the fuel assemblies assures their compatibility with the related plant systems, in particular, it concerns the chemical and mechanical compatibility. In other words, the design assures a safe fuel handling and storage both before its use in the reactor and after its removal from the reactor, and during its transportation and storage outside the reactor unit. The design enables the fuel assembly to be safely fixed, removed from and inserted into the design positions without exceeding the tolerable stress and deformation limits. The acceptance criteria for subcriticality, cooling, and release of radioactive substances into the environment are also met during the fuel handling and storage.

The design of the reactor core, reactor coolant system, system of heat removal from the primary circuit, and control and safety systems guarantees at an adequate level of confidence that the fuel elements and fuel assemblies will remain undamaged under any operation condition and that the basic safety functions will also be assured under accident conditions, including design extension conditions, except for adequate heat removal during severe accidents considered by the design.

Assessment of the preliminary concept of the reactivity control assemblies and other reactor core components

The design of the reactivity control assemblies, the absorbing materials used, the total number of absorbing elements, and the arrangement and allocation of the assemblies into functional groups and distribution across the reactor core ensure that the basic design requirements for the amount and insertion rate of reactivity are met both in the operation conditions and during design basis accidents or the design extension conditions. Meeting these requirements to a large extent depends on the fulfilment of the design requirements for the fuel elements and fuel assemblies as discussed in the part dealing with the preliminary concept of the fuel assembly, and also on the fulfilment of the design requirements for the reactor internals and for the reactivity control assembly drive mechanisms as discussed in the relevant subsections of Section 3.4. The present section focuses on aspects that are directly linked to the preliminary concept of the reactivity control assemblies and other reactor core components.

The reactivity control assemblies serve to introduce negative reactivity adequate to overcome the positive reactivity power effect due to the transition from the rated power to a zero power in the hot state, including an adequate safety reactivity margin in accordance with the requirements of the assumptions of the safety analyses for the operational conditions and the design basis accidents. This requirement is ensured for the initial conditions corresponding to any operation state of the reactor, including the period just before end of cycle, with the most negative moderator reactivity coefficient, and the conservative assumption of a single failure, e.g. with one control assembly completely withdrawn to the upper end position. The safety reactivity margin along with the design boron concentrate addition to the primary coolant shall maintain the reactor subcritical in any operation state while meeting the acceptance criteria for the design basis accidents.

The differential reactivity weight of the reactivity control assemblies and the rate of their withdrawal are limited by the design to assure that insertion of positive reactivity during undesired withdrawal of the reactivity control assemblies, i.e. during anticipated operational occurrences, should not cause exceeding of the acceptance criteria for the maximum linear power of a fuel element and DNBR.

The total reactivity weight of the reactivity control assembly is limited by the design so that the withdrawal or the ejection of a reactivity control assembly should not cause damage to the pressure boundary of the primary circuit or damage to the fuel system or reactor internals to an extent that it might cause insufficient reactor core cooling.

Burnable absorbers (both discrete and integrated) contribute to an acceptable power distribution across the reactor core, thereby reducing the risk of exceeding the acceptance criteria for the total fuel element power and hence, also DNBR. Moreover, burnable absorbers limit the boric acid concentration in the coolant at the beginning of the cycle thus ensuring that a positive moderator temperature coefficient of reactivity will not be achieved in any normal operation state of the reactor.

The design of the reactor system assures for continuous neutron flux monitoring in all design states, including the reactor loaded with fresh fuel as well as during approach to the critical state. Combination of the neutron flux measurement technology and natural and/or additional neutron sources in the reactor core, e.g. from primary or secondary neutron sources, will ensure adequate control of the core subcriticality during refuelling and approaching to the critical state.

The arrangement and design of the fuel and reactor core components assure an adequate coolant flow around the fuel elements and thus their adequate cooling. In case of need, the empty channels of the fuel assembly guide tubes are filled with hydraulic plugs which prevent excessive coolant bypass outside the virtual channels around the fuel elements. Their use results from the needs of the reactor core thermal hydraulic design.

Like with the reactivity control assemblies, the design and materials of the discrete burnable absorbers, primary and secondary neutron sources, and hydraulic plugs are such that they prevent their damage (e.g. distortion or loss of tightness) under any operation state, and prevent such damage (e.g. melting) during a design basis accident as would result in violation of the basic design requirements for subcriticality, reactor core coolability, and prevention of excessive release of radioactive substances into the environment, except for adequate heat removal during severe accidents considered by the design.

Partial preliminary assessment

The preliminary concept of the design summarising the most important requirements for the fuel system specified in the present section creates preconditions for meeting the relevant requirements of Decree No. 195/1999 Coll. [L. 266], IAEA SSR 2/1 [L. 252] and SÚJB Safety Guide BN-JB-1.0 [L. 276].

3.4.2.3 NUCLEAR CHARACTERISTICS IN THE PRELIMINARY CONCEPT OF THE DESIGN

This section describes and briefly assesses nuclear characteristics in the preliminary concept of the design. The nuclear characteristics of the design were identified based on the technical part of the tender documentation stipulating the requirements for safety and technical design of the future power plant. The assessment uses principles and approaches which are routinely applied to similar projects.

The specific design solution of the arrangement of the fuel system components within the reactor core (the "nuclear design") is described by the neutron physics parameters, which must not violate the limits ensuring compliance with the basic safety requirements and the corresponding acceptance criteria.

The nuclear design of the fuel system is defined, e.g., by the arrangement of the nuclear material and of the structural components within the fuel assembly and by the arrangement of the various fuel assembly types and other components within the reactor core.

The selected design solution shall make use of the routine potential of the nuclear design of an advanced fuel system such as: use of the spectrum of nuclear material enrichment with uranium 235, i.e. from the natural concentration to the specified maximum allowable concentration; location of the material of a given enrichment within the volume of the fuel elements / fuel assemblies; use of different burnable neutron absorber types located on the surface or inside the fuel pellet matrix; use of the spectrum of burnable absorber enrichment with a specific active isotope; burnable absorber distribution within the volume of the fuel elements / fuel assemblies; design of the other reactor core components; use of active materials of the reactor core components and their distribution within the volume of each component; arrangement of the reactor core components into specific functional units; arrangement of the fuel assemblies and other components within the reactor core; etc.

All of the above options for the design solution of the specific fuel loading within a specific fuel management system create space for setting the reactor core neutron physics parameters and, at the same time, for economical use of the nuclear material.

At the next level of this safety assessment of the design, the chosen nuclear design, including its design choices, shall be described, and its compliance with the basic design principles applicable to the nuclear fuel's nuclear design shall be assessed along with the reactivity control system.

The fuel system's nuclear design shall meet the basic safety principles specified in Section 3.3.1.

The basic safety principles are detailed into the design requirements. For the selected fuel system, the design requirements shall be detailed into acceptance criteria at the level of selected neutron physics parameters.

At the present stage of assessment of the preliminary concept of the design, we shall reduce our assessment to the key areas of the design requirements for each neutron physics parameter of the fuel system's nuclear design.

Assessment of the preliminary concept of the fuel system's nuclear design

The fuel system's nuclear design (i.e. the "design of the reactor core loading" or in its shorter form the "reactor core design") along with the fuel system design, assures meeting the basic safety functions, i.e. bringing the reactor to the subcritical state, adequate heat removal from the reactor core, and mitigation of the radiation consequences in any design basis state, including design basis accidents, or design extension conditions, except for adequate heat removal during severe accidents considered in the design. In other words, there will not be such fuel element or reactor core component damage or fuel assembly deformation which could prevent from the insertion of the mechanical control clusters into the reactor core, reduce coolant flow to below a predetermined limit, and/or disturb the design function of other components inside or outside of the reactor.

At the same time, the reactor core loading is designed so that no damage of the fuel system should occur in normal or abnormal operation states, i.e. the acceptance criteria for its damage should not be exceeded.

The importance and role of the fuel system design in meeting the above design requirements are detailed in Section 3.4.2.2.

The present section assesses in more detail the importance and role of the fuel system's nuclear design (design of the reactor core loading) and of the associated neutron-physics parameters in meeting the basic safety functions and the acceptance criteria for the fuel system damage.

Fuel burnup

Fuel burnup, which represents the total generated power in the fuel, is an appropriate parameter describing the extent of irradiation of the fuel and of the structural elements and hence, the magnitude of changes in the parameters describing the various phenomena considered for when designing the fuel system.

The design length of the fuel campaign is described by the total burnup and is basically limited by the initial amount of positive reactivity in the various fuel assembly types and by the distribution of the fuel assembly types in the reactor core.

The initial positive reactivity of the fuel assemblies constituting the design fuel loading must be adequately high to maintain the reactor core in operation at the rated power for the entire length of the planned campaign, taking into account the effect of all fission products. Excess reactivity is compensated by the burnable absorbers, reactivity control clusters, and boric acid dissolved in the coolant. The end of the campaign is usually defined by a condition when the boric acid concentration in the coolant is nearly zero and the axial position of the control clusters corresponds to the nominal operation needs. Although no specific design limitation exists for the initial positive reactivity level of the whole reactor core, there other limitations on parameters, e.g. for the reactivity coefficients and for the safety shutdown reactivity margin, which are restricting the initial positive reactivity level indirectly.

Reactivity coefficients

The fuel temperature coefficient of reactivity, referred to as the Doppler effect, is always negative for the fuel with low enriched uranium, and the moderator temperature coefficient of reactivity is basically also maintained negative in all operation states of the hot reactor due to the chosen fuel and coolant compositions. The resulting total reactivity feedback effect of the system is negative as well. A negative moderator coefficient of reactivity can be achieved by the right choice of the amount and distribution of the integral or discrete burnable absorbers or of the axial position of the control clusters. Increased amount of the reactivity compensated by the absorbers not located in the coolant can be used to reduce the importance of the moderator density change effect , in other words effect of concentration of vapour cavities on the moderator coefficient of reactivity. No specific criterion exists on the required amount of burnable absorbers. Nevertheless, the reactor core design must ensure the required design values of the moderator temperature coefficient of reactivity and the required design power distribution within the reactor core during the whole fuel campaign. The amount and distribution of the burnable absorbers in the design of the given fuel loading contribute significantly to compliance with the acceptance criteria.

Power distribution control

The reactor core design meets the initial requirements for the fuel system's nuclear design in the area of power distribution control, such as the following:

- Local power density in the reactor core must not reach a level at which the fuel would melt under the conditions of any operation state, including states with the maximum power increase
- The maximum linear power of a fuel rod must not reach a level at which the DNB might occur with sufficient confidence, i.e. the design value of DNBR be exceeded, for any operation state, including states with the maximum power increase

- The reactor core design, including fuel allocation in the reactor, must ensure that all acceptance criteria for the fuel system, as discussed in Section 3.4.2.2, are met.

Power distribution calculations for the analysed operation states include extreme conditions and power distribution shapes with all the adequate conservatisms and uncertainties, despite of a good agreement between the calculated and observed values has been demonstrated.

The reactor facility design is equipped with a power distribution monitoring system in order to enable compliance with the acceptance criteria for all operation states to be verified and ensured. The requirements for the maximum permitted levels of the specific power distribution parameters are included in the limitations and conditions for a safe operation.

Maximum controlled insertion of reactivity

The maximum insertion of reactivity due to the withdrawal of a control cluster / a group of control clusters or by the boric acid concentration reduction in the coolant is limited by the design for any operation state so that the design limits for the maximum local linear power and DNBR should not be exceeded.

Design accidents such as ejection of a cluster or break of a steam line shall not cause insertion of positive reactivity such as would prevent from a safe reactor shutdown (trip) or maintaining a fuel system geometry enabling adequate heat removal.

At the same time, the maximum insertion of reactivity due to withdrawal of a control cluster / group of control clusters shall not cause damage of the pressure boundary of the primary circuit or damage of the reactor internals such as would restrict cooling.

The maximum insertion of reactivity due to controlled or uncontrolled withdrawal of control clusters is generally limited by the maximum withdrawal speed and by the differential reactivity weight of the control clusters. Insertion of reactivity is calculated conservatively for the most unfavourable conditions of the axial power distribution and xenon concentration. Furthermore, the maximum reactivity contribution due to natural xenon decay is significantly lower than the limiting reactivity contributions caused by the control clusters.

Safety margin of subcriticality following shutdown (trip)

A specific safety margin of subcriticality must be maintained in any operation conditions. The design value of the safety margin of subcriticality following shutdown is specified, for the various operation conditions, in the limitations and conditions for safe operation, and is taken into account as one of the initial conditions in safety analyses of design basis events. The analyses of states following reactor trip include the principle of a single failure; specifically, one control cluster is assumed to be withdrawn from the reactor core.

The reactor facility design includes 2 independent reactivity control systems: (i) control clusters, and (ii) boric acid concentration in the coolant. Changes in the control cluster positions are made in order to compensate for the reactivity effects caused by changes in the fuel and coolant temperatures associated with the change of the reactor power within the span from the rated power to a hot zero power. Moreover, the control clusters are capable to provide the required subcriticality safety margin promptly after shutdown in any operation state and in accident situations, so that the acceptance criteria for the permitted extent of fuel system damage would not be exceeded.

The boric acid injection systems are able, by changing the boric acid concentration, to compensate for any reactivity changes arising from xenon burnup and from coolant density changes, thereby contributing to a safe reactor shutdown and its maintaining in the cold state.

Subcriticality of the fuel system

With the open reactor containing fuel assemblies, the boric acid concentration in coolant is maintained at a level ensuring compliance with the acceptance criteria for subcriticality. Subcriticality is maintained at a level ensuring that the critical state cannot be reached even if all the control clusters are withdrawn from the reactor core.

System provisions are made and administrative measures are taken in order to prevent uncontrolled boric acid concentration decrease. Analyses demonstrating that the fuel system would not be damaged when the mentioned provisions and measures fail have been performed.

The fuel handling and storage systems are designed so that the acceptance criteria for subcriticality without considering the control clusters and for the optimum moderation conditions are met.

The requirements for safe fuel storage and handling are included in the limitations and conditions for safe operation.

Power distribution stability

The reactor core is inherently stable with respect to power oscillations in their basic mode. Total power oscillations are reliably detected by the detectors of both the control and protective systems, irrespective of the cause of the oscillations. If the control system fails to respond to the power changes effectively enough, the protective system intervenes and ensures compliance with the applicable acceptance criteria with an adequate margin.

If spatial power oscillations occur in the reactor core during constant-power operation, they can be reliably and clearly detected and subsequently suppressed by the action of the respective control systems. The reactor core is designed so that azimuthal and radial oscillations are coherent and spontaneously and rapidly die away.

Under certain circumstances, axial spatial power oscillations could occur in the reactor core during operation. Such oscillations are reliably monitored and evaluated and subsequently suppressed by the positioning of the control clusters.

Power distribution in the reactor core is monitored by both in-reactor and out-of-reactor measurement systems, providing input information for the respective information, control and protective systems.

Partial preliminary assessment

The preliminary concept of the design summarising the most important requirements for the fuel system specified in the present section creates preconditions for meeting the relevant requirements of Decree No. 195/1999 Coll. [L. 266], IAEA SSR 2/1 [L. 252] and SÚJB Safety Guide BN-JB-1.0 [L. 276].

3.4.2.4 THERMAL AND HYDRAULIC CHARACTERISTICS IN THE PRELIMINARY CONCEPT OF THE DESIGN

The present section describes requirements for the thermal and hydraulic characteristics in the preliminary concept of the design. The properties of the design were identified based on the technical part of the reference documentation setting out requirements for the safety and technological aspects of the design of the future power plant.

The fuel shall be operated deep below the design limits in normal operation situations.

At least with a 95% probability at the 95% confidence level, critical conditions of heat transfer shall not occur at the limiting fuel rods in any conditions of normal or abnormal operation.

At least with a 95% probability at the 95% confidence level, the fuel pellet melting temperature shall not be exceeded in any conditions of normal or abnormal operation.

The number of fuel rods undergoing boiling crisis shall not exceed 10% during design basis accidents. Fuel geometry adequate for efficient reactor cooling and reactor trip shall be maintained during any design basis accident.

Partial preliminary assessment

The preliminary concept of the design summarising the key requirements for the thermal and hydraulic properties, as described in this Section "3.4.2.4 Thermal and hydraulic characteristics in the preliminary concept of the design" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Articles 8(1) and 13(1, 2).

3.4.2.5 REQUIREMENTS FOR MATERIALS FOR THE PURPOSE OF THE PRELIMINARY ASSESSMENT

This section presents a basic summary of the preliminary concept of the reactor facility design with respect to the requirements for the materials. The Preliminary Safety Report (which will follow) shall set out the conditions and procedures for the selection of suitable materials or brands of materials admissible for use in the manufacture of the nuclear plant components or of the whole unit, and basic requirements for the extent of its certification shall be specified.

Only such materials shall be permitted for use and for the manufacture of nuclear power equipment as are included in the list of steel material brands specified by applicable standards – ČSN, EN and recognised national standards or technical standards, specifications, and standards applicable to the manufacture and testing of equipment for the nuclear power sector. The materials must be approved and permitted for use in the construction/operation of a nuclear plant.

This applies to the basic materials as well as to materials that may be used as substitutes for the basic materials or that will or may be used for manufacturing, repairs or reconstructions.

Only and exclusively such materials shall be used in the manufacture of the reactor itself and of all equipment of the primary coolant circuit as have proved to be suitable for this purpose and comply with applicable rules and codes, technical standards or technical specification.

Adequate dimensioning of the components shall be demonstrated by a theoretical calculation and experimental testing. The calculation shall include a margin for the degradation of the material properties that may occur during operation due to erosion, corrosion, material fatigue, chemical environment/medium, irradiation, and ageing, as well as a margin for the uncertainty of the determination of the initial condition of the components and rate of degradation of their properties.

The design of the facility and selection of suitable materials shall ensure and guarantee that they will not be damaged due to:

- excessive deformation
- plastic instability
- elastic or elastic-plastic instability
- progressive deformation
- fatigue (progressive cracking)
- fast fracture
- corrosion-erosion damage
- thermal ageing
- radiation ageing

The design of the facility must also take into account all the possible and conceivable combinations of the various loads (temperature, pressure, radiation, ...).

Partial preliminary assessment

The preliminary concept of the design summarising the key requirements for the materials, as described in this Section "3.4.2.5 Requirements for materials in the preliminary concept of the design" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 22 paragraphs 2a, 2c; IAEA SSR 2/1 [L. 252] Requirement 47; and SÚJB Safety Guide BN-JB-1.0 [L. 276] (52), (55), (70).

3.4.2.6 PROPERTIES OF THE REACTIVITY CONTROL SYSTEMS IN THE PRELIMINARY CONCEPT OF THE DESIGN

This section describes the properties of the reactivity control systems in the preliminary concept of the design. The properties of the design were identified based on the technical part of the reference documentation setting out requirements for the safety and technological aspects of the design of the future power plant.

Reactivity shall be controlled through the following subfunctions:

- neutron moderation, absorption and reflection
- reactor core power distribution control
- keeping the reactor subcritical during outage even if one control cluster with the highest weight remains withdrawn from the core
- making up for slow reactivity changes during expected power changes

- detection of poor fuelling
- measurement of reactivity, control cluster positions, and boric acid concentration
- capability of adding boric acid in operational and accident states, keeping the facility subcritical during nuclear fuel handling and storage

Core reactivity control will be accomplished through two independent systems working on basically different design principles.

At least one of the systems shall be able to maintain subcriticality in the cool reactor during outage.

The reactivity control systems shall be designed so as to perform their safety functions in any operating state as well as in accident conditions.

During normal operation, including full-power operation, either of the reactivity control systems shall be able to shut the reactor down to a subcritical state, even if one control cluster with the highest weight is withdrawn from the core.

Adequate subcriticality (min. 1% if one control cluster with the highest weight remains withdrawn from the reactor core) shall be maintained during hot outage.

Adequate subcriticality (min. 1% with the primary circuit closed and min. 5% with the primary circuit open) shall be maintained during cold outage, either by inserting control clusters (even if one control cluster with the highest weight remains withdrawn from the core) or by increasing boric acid concentration.

The amount and rate of introduction of positive reactivity shall be limited by means of the reactor protection system to a level ensuring that postulated accidents associated with reactivity change (e.g. control cluster ejection) do not bring about non-compliance with the acceptance criteria for fuel, damage to the integrity of the primary circuit, or loss of adequate reactor core cooling.

Decreasing boric acid concentration to zero (as reasonably as possible) during normal operation shall not bring about fuel damage or damage to the primary circuit integrity, nor shall the ability to adequately cool the reactor core be reduced.No additional single failure shall be considered for this event, i.e. the event shall be analysed assuming that all control clusters are inserted.

Partial preliminary assessment

The preliminary concept of the design summarising the key requirements for the properties of the reactivity control systems, as described in this Section "3.4.2.6 Properties of the reactivity control systems in the preliminary concept of the design" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 21 and SÚJB Safety Guide BN-JB-1.0 [L. 276] (60).

3.4.3 COMPREHENSIVE PRELIMINARY ASSESSMENT OF THE CONCEPT OF THE REACTOR DESIGN

The design of ETE3,4 shall meet the safety requirements specified in Section 3.3.1.1 "Basic legislative requirements for provision of safety", as well as the requirements specified in Sections 3.3.2 – 3.3.13, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and related implementing decrees as regards nuclear safety, radiation protection and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270] documents.

To meet those requirements, the ETE3,4 design shall use reactor systems defined in the sense of Section 3.3.2.1, such as will ensure (with adequate reliability and resistance) qualitatively and quantitatively adequate technological functions in accordance with the specified requirements for the prescribed safety functions.

The principles of the design solution for the reactor facility described in Section "3.4.2 Properties of the reactor system design for the needs of the preliminary assessment" were set up based on the requirements placed by the licence applicant on the potential suppliers of the nuclear installation within the tender, and make up the concept of the design solution for this part of the design. The partial assessments performed give evidence that the expected design of the reactor facility creates preconditions for compliance with the relevant requirements for the safety and technological functions laid down by Decree No. 195/1999 Coll. [L. 266], SÚJB Safety Guide BN-JB-1.0 [L. 276], and IAEA SSR 2/1 [L. 252] and WENRA [L. 27] documents.

The particular technical solution chosen shall be specified in detail in the NPP design proper.

3.5 REACTOR COOLANT SYSTEM AND RELATED SYSTEMS

This comprehensive part of the Initial Safety Analysis Report is structured into three basic sections:

The introductory Section 3.5.1 summarises, analyses, and specifies basic legislative requirements for the reactor coolant system including its description and requirements for pressure system integrity, and for the design of the reactor pressure vessel.

Section 3.5.2 contains a description of and basic requirements for the performance of the reactor coolant system specified in the introductory Section 3.5.1 in a form that summarises the applied design requirements for relevant subsystems of the reactor coolant system in relation to all the relevant projects involved in the tender. The objective of the section is to formulate the general characteristics of the project for the purposes of the partial preliminary procedure.

The final Section 3.5.3 contains a comprehensive preliminary assessment of the concept of the reactor coolant system design, summarising the conclusions of the partial preliminary assessments performed within Section 3.5.2. It includes preliminary assessment of the concept of the design as required by the law. At the next stage of the licensing documentation the applicant shall provide evaluating and supporting information on the selected design that will make it possible to assess the reactor coolant systems' ability to provide the specified safety functions during the nuclear unit's whole lifetime in any operating state and in accident conditions. The section shall also be supplemented with detailed information at the depth and in the structure described in RG 1.206 [L. 275].

3.5.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE REACTOR COOLANT SYSTEM AND RELATED SYSTEMS

3.5.1.1 BASIC REQUIREMENTS FOR THE SYSTEM FUNCTIONS

This section describes basic legislative requirements for the system design including the primary coolant system.

At the next stage of the licensing documentation, this section shall contain a flow layout of the primary circuit and layouts of the piping and instrumentation, including functional and disposition layouts at a depth and in the structure required by RG 1.206 [L. 275].

Basic requirements

References: Decree No. 195/1999 Coll., Article 22(1)

The primary circuit and its auxiliary, control and protective systems shall be designed so that:

- a) the required strength, service life, and reliability of performance of their parts and equipment are ensured with an adequate margin in both normal and abnormal operating conditions
- b) no inadmissible coolant leaks take place

- c) they are adequately resistant against the development of failures, and any failures that may occur shall be slowed down and promptly detected
- d) the occurrence of large-scale failures is impossible
- e) action of the pressure-reducing equipment (safety valves) does not bring about inadmissible radioactivity leak from the nuclear installation
- f) primary circuit components that contain the coolant, such as the reactor pressure vessel, pressure piping, pipes and their coupling, valves and sealing and washers, including their fastening, withstand static as well as dynamic stresses expected during any operating state and in accident conditions

References: SÚJB Safety Guide BN-JB-1.0 (67)

The pressure and coolant circuit shall be designed, manufactured and tested at a quality adequate to their safety functions.

3.5.1.2 PRESSURE SYSTEM INTEGRITY

This section defines the design principles and basic requirements for the design of the heat removal system and specifies requirements for the acceptance criteria and tests relating to the pressure system integrity during the entire planned plant lifetime. At the next stage of the licensing documentation, this section shall contain an overview of the codes and standards applied to the designing and manufacturing processes, technical documentation of each component, technical specifications of the supplier, provisions for protection against extreme pressure increases, and requirements for the materials and for performance tests and inspections of the reactor coolant system's pressure boundaries. The next part will address detection of leaks from the primary circuit.

References: Decree No. 195/1999 Coll., Article 22(1a, 1c, 1d, 1f)

The primary circuit and its auxiliary, control and protective systems shall be designed so that:

- a) the required strength, service life, and reliability of performance of their parts and equipment are ensured with an adequate margin in both normal and abnormal operating conditions
- c) they are adequately resistant against the development of failures, and any failures that may occur shall be slowed down and promptly detected
- d) the occurrence of large-scale failures is impossible
- f) primary circuit components that contain the coolant, such as the reactor pressure vessel, pressure piping, pipes and their coupling, valves and sealing and washers, including their fastening, withstand static as well as dynamic stresses expected during any operating state and in accident conditions

References: Decree No. 195/1999 Coll., Article 22(2)

The proposal for the design of the primary circuit systems shall:

- a) specify such materials as have been proven for these purposes and comply with applicable codes, technical standards and/or technical specifications

- b) contain a theoretical calculation and reference to experimental testing documenting that their dimensions are adequate
- c) include a margin accounting for degradation of the properties of the material that may take place during operation due to erosion, corrosion, material fatigue, chemical environment/medium, irradiation, and ageing, and a margin accounting for the uncertainty of the identification of the initial condition of the components and rate of degradation of their properties
- d) include analysis of the limiting states with respect to the development of integrity failures
- e) specify the procedure for demonstrating the quality of manufacture and assembly by available state-of-the-art methods and for demonstrating the required tightness
- f) specify the programme and method of its in-service inspection

References: Decree No. 195/1999 Coll., Article 22(4), IAEA SSR 2/1 Req. 48

The proposed design of the primary circuit system shall keep the amount and/or pressure of coolant at levels ensuring that the design limits will not be exceeded in any normal or abnormal operating state, taking into account volume changes and leaks.

References: SÚJB Safety Guide BN-JB-1.0 (70)

The design shall include means for prompt detection of any coolant leak and enable periodic inspections and tests of the reactor's pressure and coolant circuits, including assessment of the reactor pressure vessel material.

References: WENRA App. E 2.2, 7.3, Issue K 3.1, 3.8, 3.9

The design solution shall comply with the requirements set out in Section 3.5.1.2 Barriers against radioactivity leaks and safety functions.
The design shall specify criteria for protection of the primary circuit pressure boundary: maximum pressure, maximum temperatures, temperature and pressure transients and stresses and loads.
The primary circuit pressure boundary shall be tested for leaks prior to start-up following any reactor outage such as could affect leaktightness.

The primary circuit pressure boundary shall be subjected to system pressure testing before the required period of the test. The following pressure tests shall be performed:

- Pressure test prior to reactor unit start-up after any intervention disturbing the primary circuit's leaktightness.
- Pressure test following repair during which the primary circuit integrity was disturbed
- Strength test

The design of the primary coolant circuit system shall enable the system to be tested, repaired, inspected, and periodically monitored with respect to integrity during the entire lifetime of the power plant.

3.5.1.3 REACTOR PRESSURE VESSEL

This section of the Initial Safety Analysis Report summarises basic design requirements for the reactor pressure vessel, included in specific documentation.

The reactor pressure vessel ensures safe operation of the reactor. As the reactor pressure vessel is one of the most important barriers to radioactivity leaks, its integrity must be guaranteed during any regime within the design conditions. The reactor pressure vessel provides conditions for conversion of nuclear energy to thermal energy and removal of thermal energy from the reactor core during any regime of the reactor unit during which this is required. At the next stage of the licensing documentation, this section shall contain specifications of the reactor pressure vessel materials, calculation procedures, non-destructive inspection methods, fracture toughness parameters, surveillance sample programme, and connections of the reactor pressure vessel parts. Furthermore, the section shall include analyses of the limiting pressure and temperature levels, with assessments of thermal shocks and provisions to ensure reactor pressure vessel integrity.

References: Decree No. 195/1999 Coll., Article 22(1f)

The primary circuit and its auxiliary, control and protective systems shall be designed so that primary circuit components that contain the coolant, such as the reactor pressure vessel, pressure piping, pipes and their couplings, valves, and sealing and washers, including their fastening, should withstand static as well as dynamic stress expected during any operating state as well as in accident conditions.

References: Decree No. 195/1999 Coll., Article 22(2)

The reactor pressure vessel design shall:

- a) specify such materials as have been proven for these purposes and comply with applicable codes, technical standards and/or technical specifications
- b) contain a theoretical calculation and reference to experimental testing documenting that their dimensions are adequate
- c) include a margin accounting for degradation of the properties of the material that may take place during operation due to erosion, corrosion, material fatigue, chemical environment/medium, irradiation, and ageing, and a margin accounting for the uncertainty of the identification of the initial condition of the components and rate of degradation of their properties
- d) include analysis of the limiting states with respect to the development of integrity failures
- e) specify the procedure for demonstrating the quality of manufacture and assembly by available state-of-the-art methods and for demonstrating the required tightness
- f) specify the programme and method of its in-service inspection

References: Decree No. 195/1999 Coll., Article 22(3)

The design of the primary coolant circuit shall specify conditions and requirements for its testing and maintenance, conditions of normal and abnormal operation, accident conditions, and analysis of and solutions to all influences bringing about its damage.

References: SÚJB Safety Guide BN-JB-1.0 (68)

Failures, leaks, and fast-propagating defects or breaks shall be virtually impossible as regards the reactor pressure vessel and shall not bring about design extension conditions if occurring in other components.

Resistance to brittle fracture shall be assured with an adequate safety margin in case of pressure shock during operation, maintenance, or testing or during postulated initiating events.

3.5.1.4 DESIGN OF THE SUBSYSTEMS AND COMPONENTS OF THE REACTOR COOLANT SYSTEM

This section of the Initial Safety Analysis Report summarises basic design requirements for the components and subsystems of the reactor coolant system, included in specific documentation.

At the next stage of the licensing documentation, this section shall include the design concept, description, tests and inspections, and safety assessments of the main coolant pumps, steam generators, piping, isolation systems, residual heat removal system, coolant purification system, main steam line piping and feed piping, pressuriser with safety and relief valves, safety valves, and component supports. Furthermore, this documentation shall include the design concept, description, tests and inspections, and safety assessments of the primary circuit auxiliary systems.

References: Decree No. 195/1999 Coll., Article 22 (1), IAEA SSR 2/1 Req. 47 6.13, 6.14

The primary circuit and its auxiliary, control and protective systems shall be designed so that:

- a) the required strength, service life, and reliability of performance of their parts and equipment are ensured with an adequate margin in both normal and abnormal operating conditions
- b) no inadmissible coolant leaks take place
- c) they are adequately resistant against the development of failures, and any failures that may occur shall be slowed down and promptly detected
- d) the occurrence of large-scale failures is impossible
- e) action of the pressure-reducing equipment (safety valves) does not bring about inadmissible radioactivity leak from the nuclear installation
- f) primary circuit components that contain the coolant, such as the reactor pressure vessel, pressure piping, pipes and their coupling, valves and sealing and washers, including their fastening, withstand static as well as dynamic stresses expected during any operating state and in accident conditions

References: Decree No. 195/1999 Coll., Article 22(2)

The proposal for the design of the primary circuit systems shall:

- a) specify such materials as have been proven for these purposes and comply with applicable codes, technical standards and/or technical specifications

- b) contain a theoretical calculation and reference to experimental testing documenting that their dimensions are adequate
- c) include a margin accounting for degradation of the properties of the material that may take place during operation due to erosion, corrosion, material fatigue, chemical environment/medium, irradiation, and ageing, and a margin accounting for the uncertainty of the identification of the initial condition of the components and rate of degradation of their properties
- d) include analysis of the limiting states with respect to the development of integrity failures
- e) specify the procedure for demonstrating the quality of manufacture and assembly by available state-of-the-art methods and for demonstrating the required tightness
- f) specify the programme and method of its in-service inspection

References: Decree No. 195/1999 Coll., Article 22(3)

The design of the primary coolant circuit system shall specify conditions and requirements for its testing and maintenance, conditions of normal and abnormal operation, accident conditions, and analysis of and solutions to all influences bringing about its damage.

References: Decree No. 195/1999 Coll., Article 22(4), IAEA SSR 2/1 Req. 49

The proposed design of the primary circuit system shall keep the amount and/or pressure of coolant at levels ensuring that the design limits will not be exceeded in any normal or abnormal operating state, taking into account volume changes and leaks.

References: Decree No. 195/1999 Coll., Article 22(5), IAEA SSR 2/1 Req. 49

The coolant amount/pressure controlling systems shall possess a capacity (flow rate/volume) adequate to meet the above requirements and to make the coolant makeup system capable of making up for coolant leaks and volume changes during normal or abnormal operation, taking into account coolant withdrawal for purification, so that the acceptance criteria should be complied with.

References: Decree No. 195/1999 Coll., Article 23

The primary coolant circuit system shall enable inspections of its condition to be performed periodically or continuously during operation. This also applies to tests to verify nuclear safety.

The primary coolant circuit design shall include:

- a) a programme of in-service inspections and diagnosis methods
- b) criteria for evaluating the results of the inspections and tests

References: Decree No. 195/1999 Coll., Article 25

The residual heat removal system shall ensure that the design limits for the fuel elements and for the primary coolant circuit are not exceeded during reactor outage.

The residual heat removal system shall ensure adequate redundancy of its important parts, suitable interconnection, possibility to disconnect parts of the system, and detection of leaks and possibility of their capture, so that the system will also work reliably during a single failure.

References: SÚJB Safety Guide BN-JB-1.0 (69)

In order to protect the pressure and reactor coolant circuit, requirements and conditions shall be defined for the operation and for tests of the system. This implies that the effects that may damage the circuit shall be analysed, including the highest permissible levels of static and dynamic pressures, temperatures, hydraulic or mechanical loads/stresses and pressure/temperature transients. Acceptance criteria shall be defined. The design solution of the circuit itself and of its auxiliary, control and protection systems shall ensure that the criteria are met with an adequate margin in any state accounted for in the design.

References: SÚJB Safety Guide BN-JB-1.0 (71)

Equipment protecting the system against overpressure shall be included in the design as a provision to protect the pressure circuit. Performance of that equipment shall not bring about radioactivity release into the environment or into the operating areas. Justified short-term radioactivity release into dedicated systems or areas inside the containment if necessary in order to manage transients is an exception. Anticipated operational occurrences shall be managed without intervention of that equipment.

References: SÚJB Safety Guide BN-JB-1.0 (72)

The reactor coolant circuit shall be equipped with additional isolation elements at the connection piping, preventing radioactive coolant leak to beyond the pressure/coolant circuit or to the associated systems.

References: SÚJB Safety Guide BN-JB-1.0 (74)

Components located inside the pressure/coolant circuit shall be designed so as to minimise the probability of their failure and subsequent damage of other safety-related parts of the circuit due to degradation of their properties which can be expected during operation or in accident conditions.

References: IAEA SSR 2/1 Req. 47

Components of the reactor coolant system shall be designed so as to minimise the risk of error due to the quality of material and manufacture, design standards, and possibilities of performing inspections.

The design solution of the reactor coolant system shall ensure that equipment that forms part of the primary coolant circuit's pressure boundary is not operated in the brittle fracture area.

Components located within the primary circuit's pressure boundary, such as pump driving wheels and valve components, shall be designed so as to minimise probability of their failure and subsequent damage of other safety-related parts of the circuit in any operational conditions and in accident situations. Degradation of the system components' properties that can be expected during operation shall be taken into account.

3.5.2 DESCRIPTION AND PROPERTIES OF THE DESIGN OF THE REACTOR COOLANT SYSTEM AND RELATED SYSTEMS FOR THE PURPOSE OF THE PRELIMINARY ASSESSMENT

The present section contains a basic description of and requirements for the reactor coolant system in the preliminary concept of the design. The properties of the design were identified based on the technical part of the reference documentation setting out requirements for the safety and technological aspects of the design of the future power plant. This section includes partial assessment of whether the preliminary concept of the design segment in question complies with legislative requirements specified in Sections 3.5.1.1 through 3.5.1.4. The scope of this section includes identification and assessment of the general level of the requirements for the functions of the reactor coolant systems. Particulars of the specific technological implementation shall be included in the design documentation of the selected supplier of the NPP, and the assessment will be performed within the next stage of the safety documentation.

3.5.2.1 BASIC REQUIREMENTS FOR THE SYSTEM'S FUNCTIONS IN THE PRELIMINARY CONCEPT OF THE DESIGN

The typical main components of the coolant systems of pressurised water reactors (PWRs) include: the reactor, steam generators, main coolant pumps, main circulation piping, pressuriser, pressuriser safety valve node or another system performing similar functions.

The reactor coolant system provides heat transfer from the reactor to the secondary circuit during reactor unit start-up, running, aftercooling, and shutdown. Heat generated in the reactor is transferred by the primary coolant to the steam generators to generate saturated steam, which is fed to the turbines. Chemically pure water containing an amount of boric acid serves as the coolant and, at the same time, moderator.

Pressure in the primary circuit will be maintained by a volume compensation system through coolant heating in the pressuriser or by spraying of the steam blanket in the pressuriser. Suppression of adverse pressure increase in the primary circuit will be effected not only through the flexibility of the steam blanket but also by means of a node of pressuriser safety valves or by another system fulfilling similar technological functions.

Primary coolant activity growth will be limited by coolant purification in filters.

The reactor coolant system shall be designed so as to provide reactor core cooling and reactor core heat removal to the steam generators by means of the following technological functions:

- Coolant temperature control in the core
- Coolant pressure control in the core
- Maintaining integrity of the pressure interface
- Coolant flow control through the core
- Core reactivity control
- Radioactivity confinement using the second barrier (primary circuit pressure boundary).

The reactor coolant system will use forced circulation to remove heat generated by the core through steam generators to the secondary circuit so that the primary coolant temperature is held within the required range.

This forced coolant circulation will be provided by the main coolant pumps to cover all conditions of normal operation, from cold outage to full output power.

Natural coolant circulation shall be sufficient to ensure residual heat transfer from the core to the steam generators when the reactor is shut down (full reactor coolant system, main coolant pumps off).

The reactor coolant system shall be designed so as to ensure a high degree of primary circuit integrity as one of the barriers against radioactivity leak during reactor unit operation.

The reactor coolant system shall be designed so that the basic safety requirements and principles described in Section 3.3.1, COMPLIANCE WITH BASIC REQUIREMENTS OF THE STATE SUPERVISION FOR PROVISION OF SAFETY are met. These include, in particular, the principle of defence in depth and the acceptance criteria specified for the systems, structures and components within the design basis, including the relevant design requirements based on the conditions of the site as specified in Chapter 2 and summarised for the most important ones in Section 2.10.

Partial preliminary assessment

The preliminary concept of the design summarising the most important basic requirements for the system's functions specified in the present Section "3.5.2.1 Basic requirements for the system's functions in the preliminary design" creates preconditions for compliance with the requirements stipulated by Decree No. 195/1999 Coll. [L. 266] Article 22(1) and by SÚJB Safety Guide BN-JB-1.0 [L. 276] (67).

3.5.2.2 PRESSURE SYSTEM INTEGRITY IN THE PRELIMINARY CONCEPT OF THE DESIGN

The primary coolant circuit shall provide a high-strength boundary (second barrier within defence in depth), to efficiently prevent primary coolant leakage and appreciable leaks of dissolved and/or non-dissolved radioactive substances to the containment and/or to other systems.

Primary coolant integrity shall be maintained through appropriate coolant parameter control and protection against overpressure.

This integrity shall be monitored continuously through radioactivity measurements inside the containment and in the associated systems and through leak monitoring.

The boundary between the primary circuit and the other systems will include two or more valves in series.

Those isolation valves shall be installed as close to the primary circuit piping as possible.

In some cases their function shall be combined with the containment isolating function. In such cases, one valve shall be installed inside the containment and the other outside the containment, as close to the containment wall as possible.

The operator shall have available sufficient data to separate the systems constituting the primary coolant system boundary manually.

In order to attain a high degree of tightness of the main coolant pumps, a system of filling and collection of sealing water shall be installed for the pump shaft.

Partial preliminary assessment

The preliminary concept of the design summarising the most important requirements for pressure system integrity specified in the present section 3.5.2.2, Pressure system integrity in the preliminary design, creates preconditions for compliance with the requirements stipulated by Decree No. 195/1999 Coll. [L. 266] Article 22 paragraphs 1a), 1c), 1d) and 1f) and Article 22 paragraphs 2 and 4, SÚJB Safety Guide BN-JB-1.0 [L. 276] (70), and WENRA [L. 27] App. E 2.2 and Issue K 3.1, 3.8 and 3.9.

3.5.2.3 REACTOR PRESSURE VESSEL IN THE PRELIMINARY CONCEPT OF THE DESIGN

Reactor pressure vessel

The reactor pressure vessel is a pressure vessel consisting of the vessel body and head, reactor internals located inside the vessel (e.g. core shaft, neutron reflector, etc.), and control cluster drives and instrumentation installed on the reactor head.

The main functions of the reactor include the mounting of the core, providing an adequate amount of moderator (also serving as the coolant in PWRs) required to maintain the chain fission reaction in the core, and keeping the primary coolant circuit leaktight.

The coolant enters the reactor using the inlet connections, flows through the circular gap between the vessel body and the core shaft, and penetrates to the core from below. While passing through the core, the coolant is heated by the heat released by the nuclear fuel fission reaction, and leaves the reactor through the outlet connections.

The reactor pressure vessel (RPV) shall be designed so as to fulfil the following functions:

- Provide adequate space to accommodate the reactor core and reactor internals
- Represent the reactor coolant pressure boundary
- Provide support to the reactor internals and reactor core with the aim to maintain the core and control rods in their correct positions so as to prevent excessive vibrations due to the flow
- Provide support to the control rod drives

A forging shall be used at the reactor core level. Welded joints at the location of the reactor core shall be minimised.

The number and size of welded joints on the whole RPV shall be minimised with a view to keeping the number of non-destructive tests at a minimum.

The RPV's design lifetime shall be 60 years.

The RPV shall be elastically seated in the reactor shaft, allowing for dilation.

PRV's screws and other fastening elements shall be designed so as to prevent excessive wear during planned maintenance and repairs.

All welded joints shall be accessible for in-service inspection.

Partial preliminary assessment

The preliminary concept of the design summarising the most important requirements for the reactor pressure vessel specified in the present section "3.5.2.3 Reactor pressure vessel in the preliminary design" creates preconditions for compliance with the requirements stipulated by Decree No. 195/1999 Coll. [L. 266] Article 22(1),(2),(3) and SÚJB Safety Guide BN-JB-1.0 [L. 276] (68).

3.5.2.4 DESIGN OF THE SUBSYSTEMS AND COMPONENTS OF THE REACTOR COOLANT SYSTEM IN THE PRELIMINARY CONCEPT OF THE DESIGN

Steam generator

In a nuclear power plant with a pressurised water reactor (PWR), a steam generator serves as the heat exchanger between the primary and secondary circuits. The heated primary coolant enters the hot collector, from where it is distributed into the heat-exchange tube bundle. Upon the passage of the bundle, the coolant passes the heat to the feedwater and after cooling down it enters the cold collector. After that it enters the cold branch of the core circuit loop and back to the reactor. On the secondary part of the steam generator, saturated steam is formed from the feedwater and is transferred to the turbine.

The steam generator shall comprise a pressure vessel of horizontal or vertical design with a feedwater manifold, an emergency feedwater system (if active safety systems are used), a steam collector, and a heat exchange surface formed by tubes.

The steam generator shall be designed to perform the following functions:

- Generate steam with a moisture content not exceeding 0.25%
- Maintain integrity of the primary circuit's pressure interface (the heat exchange tubes and collector are parts of the second barrier)
- Cool the coolant during normal or abnormal operation as well as during design basis accidents, when operation of the emergency feedwater system is required, and in particular, if natural primary coolant circulation is required

The tube bundle shall meet the following requirements:

The heat exchange tubes shall be welded to the primary part of the tube plate/collector and subsequently expanded to eliminate gaps between the tubes and the secondary side of the tube plate/collector. The tubes shall be supported so as to minimise any damage arising from vibrations. The tube supports shall minimise wear and fretting.

A steam generator inspection and testing programme shall be developed to demonstrate that the design of all parts enables critical areas to be periodically inspected, whereby the strength and leaktightness can be demonstrated.

Main coolant pump

The main coolant pumps shall provide circulation of the required amount of the coolant in the primary circuit matching the heat output of the reactor in the various operating regimes.

The main coolant pump (MCP) shall have a sufficient run to the rest to provide adequate flow through the reactor core in the event of loss of power at the pumps. In other words, the MCP shall have a sufficient moment of inertia.

Proven sealing, thermal shielding and pump component solutions shall be applied. The results of their tests shall be submitted unless the designs have been tested by operation.

In the event of MCP outage, leaks via the shaft sealing shall also be limited in the event of failure of the auxiliary systems. This does not apply to encased MCPs.

Pressuriser system

The pressuriser system consists of the pressuriser, bubble tank, safety valve nodes and the tubing connecting the individual devices to the follow-up systems. The pressuriser assembly includes electric boilers and a sprinkling system.

The pressuriser system shall serve to make up for coolant volume changes, to maintain the pressure and to limit pressure variations in the primary circuit, to protect against uncontrolled pressure increase during emergency, as well as to provide smooth pressure increase/decrease during primary circuit heating and aftercooling. Pressure in the primary circuit shall be created and maintained by heating the pressuriser's water space or by injecting the primary coolant into the pressuriser's steam space.

A safety valve node or another system performing similar technological functions shall serve to suppress undesirable pressure increase in the primary circuit during failure of the technological equipment and safety systems.

The pressuriser shall be designed to perform the following functions:

- Primary circuit pressure control
- Absorption of primary coolant volume changes
- Maintain pressure and structural integrity of the primary circuit as a part of the primary circuit's pressure boundary

The pressuriser and safety valves shall be designed to control overpressure in the primary circuit during anticipated operational occurrences and during abnormal operation. Peak pressure level must not exceed 110% of the design pressure.

Normal residual heat removal system

The normal residual heat removal system shall serve to adequately remove heat from the reactor coolant system (primary coolant circuit) in conditions in which heat removal via steam generators is no more efficient. The system shall remove heat from the reactor coolant system continuously during outages, including operating modes during refuelling.

The normal residual heat removal system shall be designed to perform the following functions:

- Maintain temperature in the primary circuit during refuelling. maintenance work and other work performed during reactor outage
- Control the primary coolant flow while the primary circuit is in operation and the MCPs are shut down

- Maintain integrity of the primary circuit's pressure boundary always when the system is connected to the primary circuit
- Protect the primary circuit protection against high pressure at low temperatures

Detailed requirements for the residual heat removal system are specified in Section "3.6.2.3.5 Residual heat removal system in the preliminary concept of the design".

Partial preliminary assessment

The preliminary concept of the design summarising the most important requirements for the subsystems and components of the reactor coolant system specified in the present section "3.5.2.4 Design of the subsystems and components of the reactor coolant system in the preliminary design" creates preconditions for compliance with the requirements stipulated by Decree No. 195/1999 Coll. [L. 266] Article 22(1-5), IAEA SSR 2/1 [L. 252] Req. 47, and SÚJB Safety Guide BN-JB-1.0 [L. 276] (67), (71), (72) and (74).

3.5.3 COMPREHENSIVE PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT OF THE REACTOR COOLANT SYSTEM AND ASSOCIATED SYSTEMS

The ETE3,4 design shall meet the safety requirements specified in Section 3.3.1.1 as well as the requirements specified in Sections 3.5.1.1 to 3.5.1.4, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and related implementing decrees as regards nuclear safety, radiation protection and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270] documents.

To meet those requirements, the ETE3,4 design shall use the reactor coolant system and associated systems such as will ensure (with adequate reliability and resistance) qualitatively and quantitatively adequate technological functions in accordance with the specified requirements for the prescribed safety functions.

The principles of the design solution for the reactor coolant system and associated systems described in Section "3.5.2 Description of the design of the reactor coolant system and related systems for the needs of a preliminary assessment" were set up based on the requirements placed by the licence applicant on the potential suppliers of the nuclear installation within the tender, and make up the concept of the design solution for this part of the design. The partial assessments performed give evidence that the expected design of the reactor coolant system and associated systems creates preconditions for compliance with the relevant requirements for the safety and technological functions laid down by Decree No. 195/1999 Coll. [L. 266], SÚJB Safety Guide BN-JB-1.0 [L. 276], and IAEA SSR 2/1 [L. 252] and WENRA [L. 27] documents.

Details of the ultimate specific technological implementation of the reactor coolant system and associated systems will only be described in the NPP design.

3.6 ENGINEERED SAFETY FEATURES (SAFETY SYSTEMS)

This comprehensive part of the Initial Safety Analysis Report is structured into three basic sections:

The introductory Section 3.6.1 summarises, analyses and specifies basic legislative requirements for the most important safety systems, including their description, properties, design solution principles, inspections and tests. Included sections are containment systems, emergency core cooling system, habitability systems, and fission product removal and control systems.

Section 3.6.2, which follows, contains a description and specification of the basic requirements for the functions of the safety systems specified in the introductory section 3.6.1 in a form that summarizes the applied design requirements for the relevant subsystems of the safety systems in relation to all the relevant projects involved in the current tender procedure. The objective of the section is to formulate the general characteristics of the project for the purposes of the partial preliminary procedure.

The final Section 3.6.3 contains a comprehensive preliminary evaluation of the concept of the safety systems, summarising the conclusions of the partial preliminary assessments presented in Section 3.6.2. Within the assessment of the summary of requirements so obtained, the section contains a preliminary evaluation of the project concept as required by the law. At the next stage of the licensing documentation the applicant shall provide evaluating and supporting information on the selected project that will make it possible to assess the ability of the safety systems and associated systems to perform the specified safety functions during the whole lifetime of the reactor unit in any defined operating state and in accident conditions. The section shall also be supplemented with detailed information at the depth and in the structure described in RG 1.206 [L. 275].

3.6.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE SAFETY SYSTEMS

3.6.1.1 MATERIALS OF THE SAFETY SYSTEMS

Applicable legislation, cited in the Terms of reference, does not specify any requirements for the safety system materials. The materials shall be specified within the next stages of the licensing documentation.

3.6.1.2 CONTAINMENT SYSTEMS

This part of the Initial Safety Analysis Report summarises basic design requirements for the containment system, including the depressurization system and heat removal and the associated systems. It also deals with penetrations through the containment wall and with requirements for inspections and tests. The reactor containment system is a safety system to confine and prevent leaks of radionuclides and ionising radiation produced by the radionuclides into the environment.

At the next stage of the licensing documentation, this section shall be completed with demonstration of the containment system's ability to fulfil its safety functions to the depth of detail and in the structure set out in RG 1.206 [L. 275].

3.6.1.2.1 Basic functions

References: Decree No. 195/1999 Coll., Article 32; IAEA SSR 2/1 Req. 54-55; SÚJB Safety Guide BN-JB-1.0 (107)

The nuclear installation shall be equipped with a safety system of the reactor containment to limit leaks during normal or abnormal operation to a level as low as reasonably achievable (ALARA principle) and below the authorised limits. In design accident conditions included in design basic, associated with radionuclide leaks, the system shall reduce the leaks (and the associated ionising radiation) into the environment to levels complying with the limits specified by applicable legislation (Decree No. 307/2002 Coll. [L. 4]), unless this function is executed by other technical means.

Furthermore, the reactor containment system shall protect the reactor from natural evants and from man-made events in accordance with Section 3.3.3.

The reactor containment system shall provide radiation shielding during operating states as well as in accident situations.

3.6.1.2.2 Principles of the approach

References: Decree No. 195/1999 Coll., Article 33; WENRA App. E 9.8, 9.9, 9.10, F 4.1, 4.2, 4.3, 4.4, 4.5, 4.6, 4.7; SÚJB Safety Guide BN-JB-1.0 (108 – 117)

The containment system consists of the following items:

- Containment enclosing the hermetic zone, dimensioned for any design basis accident
- Features for izolation
- Systems for control of pressures and temperatures
- Ventilation and filtration systems
- Other auxiliary systems

Additional acceptance criteria (including limits for the temperature and pressure inside the containment and leaktightness of the containment) shall be specified to protect the containment and perform its functions. The design shall make for conditions ensuring that the criteria will not be exceeded:

- during design basis accidents, for a time which is sufficiently long to attain a safe and stable condition
- during severe accidents, for a time no shorter than as needed to implement protective measures in accordance with applicable legislation (Decree No. 318/2002 Coll. [L. 26])

The containment system shall be functional also at the highest pressures / underpressure levels and temperatures occurring during design basis accidents. Other factors that need to be considered include the effects of the systems for pressure and temperature control inside the containment, effects of other potential energy sources, penetrations and passages, inaccuracies of the calculation models, results of experiments, and operational experience.

The design shall ensure that any loss of the containment safety functions is practically eliminated, and that procedures, technical means or organisational measures shall be in place to attain the highest possible degree of protection of the containment's integrity and performance in accident conditions, including severe accidents, so that the consequences of excessive pressures, overheating, damage by explosive gases, integrity degradation by the action of the melt from the degraded remainders of the reactor core, radioactivity leaks in liquids or aerosols, reactor core melt, etc., would be minimised.

The containment system shall be designed so as to provide physical protection of the nuclear installation and nuclear material in accordance with the requirements stipulated by applicable legislation (Decree No. 144/1997 Coll. [L. 285]).

In accident conditions the containment system shall separate equipment located inside the containment's hermetic part from the remaining part of the nuclear installation by means of testable isolation elements in accordance with the requirements set out in Section 3.6.2.3.

The containment system shall be able to reduce the consequences of any identified bypass of the containment's hermetic space boundaries. Any route that passes through the containment and:

- is a part of the boundary of the reactor's pressure and cooling unit or is directly connected to the containment's atmosphere shall be reliably separable, such routes being equipped with two or more isolation elements connected in series (one inside and one outside of the containment)
- is not a part of the boundary of the reactor's pressure and cooling unit, nor is it directly connected to the containment's atmosphere, shall be fitted with a minimum of one isolation element installed outside of the containment.

References: Decree No. 195/1999 Coll., Article 40; SÚJB Safety Guide BN-JB-1.0 (113)

Adequate ventilation ducts shall exist between the parts of the containment's hermetic area so as to prevent local accumulation of explosive gases and/or damage to the containment or other equipment of the containment system due to pressure differences developing in accident situations.

References: IAEA SSR 2/1 Req. 30, 58, 6.27, 6.28, 6.29, 6.30

Coverings, insulation and coating of equipment, structures and components inside the containment system shall be selected deliberately and methods for their use shall be specified so as to support their safety functions and minimise the adverse effect of their degradation on the remaining safety systems.

Conditions of the environment that can be expected or that will arise from special operating states, such as containment system leak rate tests, shall be included in the equipment qualification programme.

3.6.1.2.2.1 Containment heat removal system

References: Decree No. 195/1999 Coll., Article 41; SÚJB Safety Guide BN-JB-1.0 (115)

The containment system shall be equipped with a safety heat removal system that:

- during and after design basis accidents will, in conjunction with other systems, sufficiently rapidly reduce pressure and temperature in the hermetic zone to the design level
- during severe accidents will ensure, in conjunction with other available systems of the nuclear installation, that the containment's tightness remains reasonable for the time needed to implement protective measures pursuant to applicable legislation (Decree No. 318/2002 Coll. [L. 26])
- will ensure reliability, redundancy and functional diversity of its important equipment, and function of the systems during a single failure.

3.6.1.2.2.2 Other equipment of the reactor containment system

References: Decree No. 195/1999 Coll., Article 42; SÚJB Safety Guide BN-JB-1.0 (116)

The containment system shall be equipped with systems to control radionuclides and substances that might penetrate into the containment in accident conditions. Such system shall, in conjunction with other systems, be able to:

- reduce volume activity and modify the composition of the fission products
- control volume concentrations of explosive materials in order to ensure integrity of the hermetic envelope and reduce the amount of leaking radionuclides

References: IAEA SSR 2/1 Req. 58, 5.30

The technical equipment of the containment system or of the entire nuclear installation shall ensure that hermetic envelope damage by the molten fuel is virtually impossible. The events to be analysed shall be selected based on engineering judgement and the results of safety analysis.

3.6.1.2.3 Passages through the containment

References: IAEA SSR 2/1 Req. 55, 6.20, 6.21; SÚJB Safety Guide BN-JB-1.0 (111)

The number of penetrations through the containment walls shall be minimised and each penetration/passage shall meet the same requirements as the containment structure.

The penetrations/passages shall be protected against the effects of the reactive forces arising from the motion of the piping or from loads arising during accidents, such as flying objects, flowing medium and pipe whips.

3.6.1.2.3.1 Penetrations through the containment

References: Decree No. 195/1999 Coll., Article 37; SÚJB Safety Guide BN-JB-1.0 (111)

Containment penetrations for pipes and cables shall be designed so that:

- any leaks can be detected, confined and collected
- they can be periodically tested for leaks at the design pressure, independently of the containment leak rate tests
- they are protected from the effects of dynamic forces arising from motions of the pipes and from loads arising during accidents, e.g from flying objects, flowing medium and pipe whips

3.6.1.2.3.2 Isolation valves

References: Decree No. 195/1999 Coll., Article 38; IAEA SSR 2/1 Req. 56, 6.22, 6.23, 6.24; SÚJB Safety Guide BN-JB-1.0 (109)

Isolation valves shall meet the following requirements:

- Pipes passing through the containment that are parts of the primary circuit boundary or are directly connected to the atmosphere in the hermetic zone shall be equipped with reliable isolation systems, each comprising two or more isolation valves arranged in series, located outside and inside the containment, independently and reliably controllable, and providing suitable leak detection. The outer isolation valves shall be located as close to the containment as possible.
- Pipes passing through the containment walls that are not parts of the primary circuit boundary, nor are they directly connected to the atmosphere inside the containment, shall be equipped with a minimum of one outer isolation valves, located as close to the containment as possible.
- Their design shall enable the isolation valves to be subjected to periodic leak testing.
- The isolation system shall perform well even in the event of a single failure beyond the isolation valve's mechanical part.
- The above requirements need not be met by some specific lines such as instrumentation lines for which it can be demonstrated that the provisions implemented for insulation of the containment are acceptable with respect to acceptance criteria defined on a different basis.
- Except for justified exceptions addressed through a different approach, the isolation valves shall be an automatically controlled isolation element or a manually closable isolation element or controllable isolation element or a isolation element with remote manual control.

3.6.1.2.3.3 Airlocks

References: Decree No. 195/1999 Coll., Article 39; SÚJB Safety Guide BN-JB-1.0 (112)

The containment system shall be equipped with means enabling personnel, in justified cases (such as for inspection) to enter the hermetic area during operation. Airlocks through the containment shall be equipped with double doors controlled so that their tightness is preserved at all times. The tightness of the airlocks shall match that of the reactor containment system.

References: IAEA SSR 2/1 Req. 57, 6.25, 6.26

If the design includes door(s) for personnel entrance for inspections, provisions to ensure personnel safety shall be specified in the design. Such requirements shall also apply to the passage systems.

The tightness of openings for conveying material and equipment shall match that of the containment system, and the possibility shall exist for such openings to be closed quickly and reliably should the requirement for separating the hermetic zone from the surroundings emerge.

3.6.1.2.4 Testing and inspections

References: SÚJB Safety Guide BN-JB-1.0 (114)

The containment system shall enable the parts to be tested for leaktightness, strength and performance in order to verify compliance with the acceptance criteria. This shall be possible after finishing the construction work as well as periodically during operation (in-service testing) and following repairs, so that any degradation of the components/systems can be identified and appropriate corrective measures applied.

3.6.1.2.4.1 Containment leak rate test

References: Decree No. 195/1999 Coll., Article 34

The containment and its leaktightness-related equipment shall be designed so as to enable:

- leaktightness testing at the design pressure after installation of all penetrations and passages
- periodic in-service containment system leaktightness testing at the design pressure or at lower pressures, so that compliance with the design requirements can be verified by extrapolation of the results.

The containment and its tightness-related equipment shall enable completed repairs to be tested for tightness at the design pressure.

3.6.1.2.4.2 Containment structure integrity test

References: Decree No. 195/1999 Coll., Article 35

Prior to putting the nuclear installation in operation it shall be possible to test the containment pressure strength, in order to demonstrate the hermetic containment's integrity at the test pressure, which shall be higher than the maximum design pressure.

3.6.1.2.4.3 Containment system in-service inspection

References: Decree No. 195/1999 Coll., Article 36

The containment system shall enable the following in-service tests and inspections to be performed:

- Periodic inspections of its various parts and equipment
- Performance tests of its various parts and equipment

References: WENRA Issue K 3.13

The containment system integrity inspection shall include:

- Integral leak rate tests
- Leak rate tests and, where appropriate, performance tests of the airlocks, penetrations, closures, and isolation valves
- Integrity tests from the structure aspects, e.g. tests of the containment system's lining and prestressing systems

3.6.1.3 EMERGENCY CORE COOLING SYSTEM

This part of the Initial Safety Analysis Report summarises basic national and selected international legislative requirements for the design of the emergency core cooling system. At the next stage of the licensing documentation, this section shall be completed with additional information on emergency core cooling system at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.6.1.3.1 Purpose and description of the system

The emergency system shall provide reliable core cooling during design basis accidents associated with loss of coolant and/or loss of normal heat removal. Its safety function rests in heat removal for a time long enough to ensure that the fuel element cladding temperatures as affected by the subsequent energy contribution from chemical reactions (cladding, water, hydrogen release) will not exceed the levels determined by the acceptance criteria for design basis accidents, and to prevent any changes in the core components and/or reactor internals affecting the cooling efficiency and reactor shutdown systems.

3.6.1.3.2 Principles of the approach

References: Decree No. 195/1999 Coll., Article 22(1); SÚJB Safety Guide BN-JB-1.0 (56, 57, 71, 75); IAEA SSR 2/1 Req. 48, 6.13, 6.14, 6.15

The nuclear installation shall be equipped with safety systems providing the basic safety functions during anticipated operational occurrences and design basis accidents. They may be systems based on passive principles or they may constitute a set of active systems designed to assure their performance for the required time with a high reliability.

The primary circuit and its auxiliary, control and protective systems shall be designed so that the following requirements are met:

- The required strength, service life, and reliability of performance of their parts and equipment is ensured with an adequate margin in both normal and abnormal operational conditions
- No inadmissible primary coolant leaks can occur, and the interconnected piping of the auxiliary and safety systems is fitted with appropriate isolation devices to limit any coolant loss and prevent design basis accidents associated with coolant loss through those interconnected systems
- They are adequately resistant against the development of failures, and if any failures occur, they slow down their development and make for a prompt detection of failures
- Any large-scale failures are eliminated; this encompasses, among others, plant operation in conditions where the primary circuit material could exhibit brittle behaviour
- They are equipped with devices protecting the primary circuit against overpressure, whose actuation (relief valves) shall not result in inadmissible radioactivity release from the nuclear installation into the environment or into the operating areas, with the exception of justified and time-limited radioactivity release into dedicated areas or areas inside the containment if necessary in order to manage transients; anticipated operational occurrences shall be managed without actuation of those devices
- Primary circuit components that contain the coolant, such as the pressure vessel, pressure piping, pipes and their coupling, valves and sealing and washers, including their fastening, withstand static as well as dynamic stresses expected during any operating state as well as in accident conditions
- In any operating state with the critical reactor, the resulting effect of the instantaneous feedback components in the reactor core counteracts any rapid reactivity increase

References: Decree No. 195/1999 Coll., Article 22(2, 6), SÚJB Safety Guide BN-JB-1.0 (82), IAEA SSR 2/1 Req. 47, 6.16

The primary coolant circuit equipment design shall meet the requirements described in Section 3.5 (Reactor cooling system and associated systems), and shall provide the operators with appropriate technical means and enable them to take organisational steps in order to prevent the development of reactor core melting at a high pressure in the primary coolant circuit in severe accident situations.

3.6.1.3.3 Emergency core cooling system

References: Decree No. 195/1999 Coll., Article 26(a, b); SÚJB Safety Guide BN-JB-1.0 (76, 77, 78); IAEA SSR 2/1, Req. 52, 6.18, 6.19

The emergency cooling system shall ensure:

- reliable cooling of the reactor core in accident conditions associated with loss of coolant (up to the equivalent of sudden break of the largest-diameter pipe of the piping system) so that:
    - the fuel element cladding temperatures will not exceed the limiting design levels
    - the energy contribution from chemical reactions (cladding, water, hydrogen release) will not exceed the admissible level
    - no changes in the fuel elements, fuel assemblies and/or reactor internals that might affect the cooling efficiency will occur
    - residual heat will be removed for an adequately long time
- ensure – through its adequate capacity, appropriate configuration of location of the connection of those systems to the pressure and cooling circuit, redundancy, functional diversity, appropriate interconnection, the possibility of disconnecting parts of the system, detection of leaks and possibility of their trapping – that the system will perform its safety functions – irrespective of whether it is supplied with power from the grid or, in accident situations, from the nuclear installation's backup system in accordance with the requirements for power supply systems (Section 3.8 Electrical systems) – reliably even during a single failure of the equipment

In situations for which it cannot be demonstrated that functional failure and simultaneous loss of integrity of any of the emergency core cooling systems is extremely unlikely, the systems shall be able to perform their design functions also during a single failure and, at the same time, in a situation where one of the parts of the system is inoperable because of maintenance.

3.6.1.3.4 In-service inspection of the emergency core cooling system

References: Decree No. 195/1999 Coll., Article 27; SÚJB Safety Guide BN-JB-1.0 (79)

The emergency core cooling system shall be designed to enable periodic tests and inspections of:

- the system strength and tightness
- the system's active elements (including performance tests)
- the emergency cooling system as a unit, including performance tests in conditions as encountered during its action (sequence of operations activating the various systems, switching to a backup power supply system, switching to an alternative cooling water system,...)

3.6.1.3.5 Residual heat removal system

References: Decree No. 195/1999 Coll., Article 25(1, 2); SÚJB Safety Guide BN-JB-1.0 (81, 118); IAEA SSR 2/1, Req. 51, 53, WENRA App. E 9.7

During reactor shutdown and during and after anticipated operational occurrences and accidents, the residual heat removal system shall (even independently of power sources beyond the nuclear installation – see Section 3.8 Electrical systems) remove heat evolved from the decay of the fission products and accumulated heat of the components and heat from nuclear safety-related systems so that the acceptance limits of the fuel and the reactor's pressure and cooling unit and nuclear-safety-related systems are not violated. Through adequate redundancy of the important components of the residual heat removal system, suitable interconnection, possibility of disconnecting parts of the system, detection of leaks and the possibility of their trapping, the system shall perform this function also during a single failure and, at the same time, in a situation where one of the parts of the system is inoperable because of maintenance. The system shall efficiently prevent personnel exposure to ionising radiation and penetration of radioactivity into the environment. To this end, the system shall be equipped with appropriate means, including means for monitoring those functions.

In order to ensure the function of the safety systems for residual heat removal from the reactor core and the containment and for heat removal from nuclear safety-related systems during normal and abnormal operation as well as in accident situations, the nuclear installation shall be equipped with an independent heat removal system reaching as far as the terminal heat sink, provided with an adequate degree of redundancy of the important systems as well as of power supply (see Section 3.8 Electrical systems), equipped with a system for detecting any ingress of radioactivity into the system and with means for preventing radioactivity leak into the environment.

3.6.1.3.6 Fire protection

References: WENRA Issue S 2.1

The emergency cooling system and the residual heat removal system shall be designed so as to minimise the frequency and impacts of fires on the safety function of those systems and enable the condition of the nuclear installation during and after fire to be monitored (Section 3.9.1.5.1 Fire protection programme).

3.6.1.4 HABITABILITY SYSTEMS OF CONTROL ROOMS

This section describes basic legislative requirements for the design of habitability systems of the control rooms.

3.6.1.4.1 Purpose and description of the system

The habitability systems of the control rooms shall provide such parameters of the environment in the control rooms (both main control room and supplementary control room) which provide adequate comfort, safety and health conditions for the personnel and maintain the installed equipment appropriately operable in any operating state as well as in accident conditions.

3.6.1.4.2 Main **control room**

References: Decree No. 195/1999 Coll., Article 20(2), IAEA SSR 2/1 Req. 65, 6.39

The habitability system of main control room shall provide such parameters of the environment as are suitable for personnels' work and for the installed equipment during any operating state as well as in accident conditions and comply with rules and regulations applicable to the workplace.

References: SÚJB Safety Guide BN-JB-1.0 (94); IAEA SSR 2/1 Req. 65, 6.40

When developing the design of the system, all internal and external events (see Sections 3.3.3, 3.3.5) which may pose a direct threat to its continued operation, and the design shall provide for reasonably practicable measures to minimize the effects of such events..

References: Decree No. 195/1999 Coll., Article 20(3)

Ergonomic approaches shall be applied to the design of the control room layout and arrangement.

3.6.1.4.3 Supplementary **control room**

References: Decree No. 195/1999 Coll., Article 20(3); SÚJB Safety Guide BN-JB-1.0 (96)

For times when the main control room is not available there shall be available for activities associated with reactor shutdown (scram) and with maintaining the reactor in a safe condition, including NPP status monitoring from supplementary control room.

The supplementary control room shall be adequately physically and electrically separated from the main control room, and the habitability systems of control rooms shall be designed so as to provide the personnel with appropriate comfort, safety and health conditions and to maintain parameters of the environment appropriate for the performance of the equipment during any operating state as well as in accident situations. The habitability of the supplementary control room shall also be assured in accident situations unless the occurrence of such situations can be practically eliminated by applying the probabilistic approach.

References: IAEA SSR 2/1 Req. 66, 6.41

When developing the design of the system, all internal and external events (see Sections 3.3.3, 3.3.5) which may pose a direct threat to its continued operation, and the design shall provide for reasonably practicable measures to minimize the effects of such events.

3.6.1.5 SYSTEMS TO CONTROL RADIOACTIVITY LEAKS IN ACCIDENT SITUATIONS

This part of the Initial Safety Analysis Report summarises basic design requirements for the systems to control (reduce) radioactivity leaks in accident situations. At the next stage of the licensing documentation, this section shall be completed with additional information at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.6.1.5.1 Purpose and description of the system

The safety systems to reduce (control) radioactivity leaks in accident situations (such as ventilation systems with air filtration, spray system) shall reliably reduce the release and/or migration of radioactive substances in the environment to a level As Low As Reasonably Achievable (ALARA), which in design basis accident situations shall be lower than the regulatory limits. Their safety function rests in reducing the volume activity and/or modifying the composition of the radionuclides in accordance with the acceptance criteria.

3.6.1.5.2 Principles of the approach

References: Decree No. 195/1999 Coll., Article 42 1); SÚJB Safety Guide BN-JB-1.0 (116); IAEA SSR 2/1 Req. 55, 58, 6.29

The containment system shall be equipped with systems to control radionuclides that might penetrate into the containment in accident situations. The systems shall be able, in conjunction with the remaining systems, to reduce volume activity of the radionuclides and to modify their composition in accordance with the acceptance criteria so that the leaks should comply with applicable legislation (Decree No. 307/2002 Coll. [L. 4]) and should be at the lowest practically achievable level.

References: Decree No. 195/1999 Coll., Article 44; SÚJB Safety Guide BN-JB-1.0 (119); IAEA SSR 2/1 Req. 81, 6.71, 6.72

The nuclear installation shall be equipped with ventilation, air conditioning, and filtration systems that, in any operating state as well as in accident situations, will maintain the prescribed conditions in areas where nuclear safety-related equipment is located. In addition, the systems shall:

- prevent any dispersion and/or uncontrolled leak of radionuclides in the various nuclear installation's areas in accordance with the requirements for their accessibility
- reduce volume activities of any radionuclides dispersed and leaked into the nuclear installation's accessible areas so that the dose limits laid down in Decree No. 307/2002 Coll. [L. 4] are not exceeded and the doses are at the lowest practically achievable level
- maintain the specified climatic conditions
- keep radionuclide leaks below the specified limits
- provide for air streaming from areas with lower contamination with harmful substances to areas with higher contamination so that the areas with higher contamination should be held in the vacuum mode
- be equipped with reliable and efficient filters and shall enable the filter efficiency to be tested
- possess adequate redundancy of important systems enabling them to work during a single failure

3.6.1.5.3 Fire protection

The requirements for fire protection of buildings and systems containing radioactive materials are specified in Section "3.9.1.5.1 Fire protection programme".

3.6.2 DESCRIPTION AND PROPERTIES OF THE DESIGN OF THE SAFETY SYSTEMS FOR THE PURPOSE OF THE PRELIMINARY ASSESSMENT

This section describes the properties of the design of the safety systems for the preliminary assessment of the design concept. The properties of the design were identified based on the technical part of the reference documentation setting out requirements for the safety and technological aspects of the design of the future power plant. This section includes partial assessment of whether the preliminary concept of the design segment in question complies with the requirements specified in Sections 3.6.1.1 to 3.6.1.5. The scope of this section includes the determination and assessment of the general level of requirements for the performance of the safety systems. The specific technological implementation of the systems shall be included in the design of the nuclear power plant and assessed in the next stage of the safety documentation.

The basic requirements for the design of the safety systems are based on the general requirement for application of the principle of defence in depth (the requirements for the concept of defence in depth are specified in Section 3.3.1.1.3 and the requirements for the process of defence in depth are specified in Section 3.3.1.1.4).

The safety systems shall be designed so as to meet the basic safety requirements and principles described in Section 3.3.1. These include, in particular, the principle of defence in depth and the acceptance criteria specified for the systems, structures and components specified within the design basis, including the relevant design requirements based on the conditions of the site as specified in Chapter 2 and summarised for the most important ones in Section 2.10.

Either of the ETE3,4 reactor units shall be a fully autonomous design that includes the safety systems and related subsystems described below. The specific approach to the technological implementation will only be included in the proposed design of the nuclear power plant. The design solution selected for implementation need not include all systems, structures, and components specified in the present section; however, if it uses any of them, they shall meet the requirements specified here.

3.6.2.1 SAFETY SYSTEM MATERIALS IN THE PRELIMINARY CONCEPT OF THE DESIGN

As regards the materials selected for the safety systems, no assessment has been performed because applicable legislation does not stipulate any requirements in this respect.

Partial preliminary assessment

As regards the materials selected for the safety systems, no assessment has been performed because applicable legislation does not stipulate any requirements in this respect.

3.6.2.2 CONTAINMENT SYSTEMS IN THE PRELIMINARY CONCEPT OF THE DESIGN

The reactor containment system is a fundamental safety system involved in the safety function to trap and control leaks of radionuclides and ionising radiation produced by them into the environment.

The containment system shall consist of an inner hermetic envelope and the outer containment. It shall include temperature and pressure control systems and systems to control (reduce) radioactivity leaks into the environment in accident situations (e.g. the passive heat removal system, spray system, hydrogen combustion system, and ventilation and filtration systems).

The hermetic envelope shall be designed so that it should, in operating states and in accident conditions associated with radioactivity leaks, including severe accidents, reduce radioactivity leaks into the environment and ensure that the radiological consequences in the plant surroundings are at an acceptable level.

The containment design shall ensure that the reactor pressure vessel, primary coolant circuit and all nuclear safety and radiation safety-related equipment located inside the hermetic envelope are protected against the external events described in Chapter 2.

The containment system shall also act as biological shielding.

3.6.2.2.1 Basic functions in the preliminary concept of the design

The design of the NPP shall include a containment system performing the following safety functions:

- Fission products confinement and control
- Equipment protection against external events
- Biological shielding

The containment system design shall assure protection of the reactor coolant pressure boundary and of the fuel damage prevention systems against external events. In case of an external event, the containment system integrity shall be maintained at a level allowing the reactor to be shut down and residual heat to be removed from the reactor core. The ability to remove residual heat shall be maintained at a level required to prevent loss of integrity of the containment and of the reactor coolant pressure boundary as well as to prevent failure of systems needed to prevent reactor core damage.

The containment system and reactor internals shall be designed so that exposure of the public and staff outside the containment from sources contained inside the containment will be minimised. The design shall assure that the radiation burden is kept within limits allowing personnel to do their work in any state of the NPP including accident situations. To enable activities necessary to manage emergencies to be performed, the areas and access routes to them shall be fitted with adequate shielding.

3.6.2.2.2 Principles of the solution in the preliminary concept of the design

The supplier shall define critical parameters for the design based on the criteria described below. The design pressure (P_{des}) and design temperature (T_{des}) shall be determined so as to meet the acceptance criteria for any combination of loads.

The test pressure shall be defined primarily based on P_{des} in accordance with applicable standards and legislative requirements.

The design levels of total leak shall be determined in dependence on the specific concept of the containment design. The solution of the containment system used in the design shall ensure that the design criteria are met during design basis accidents and for a time sufficient to implement protective measures in accident conditions.

The following steps shall be included in the design process in order to fully specify the behaviour of the containment structure in any nuclear power plant conditions (i.e. testing, design and accident conditions):

- Identification of the various loads and their combinations
- Identification of the desirable behaviour of the structure
- Definition of the acceptance criteria to assure the required behaviour of the structure

The required safety functions shall cover the maximum pressures (and vacuum levels where applicable) and temperatures of the design basis accidents.

The overall behaviour of the structure shall be evaluated with a view to identifying the potential effect of weaknesses, assessment of margins with respect to the limiting levels, and identification of potential damages to the structure. Local effects and details of the design of the components shall also be considered in order to identify the potential mechanisms.

The concept of the accepted design shall practically eliminate the hazard of loss of the containment system's functions in accident conditions. This shall be achieved by means of the technological equipment and organisational procedures preventing degradation of the system structure and integrity which might result from pressures and temperatures exceeding the respective criteria, pressure impacts due to gas explosions, damage by the action of the melt, etc.

As a minimum, the ducts shall be equipped with separating elements as follows:

- Two isolation valves (one outside, the other inside) for those ducts passing through the containment that are parts of the reactor's pressure and cooling unit or that are directly connected to the hermetic zone atmosphere
- One isolation valve (outside) for those ducts passing through the containment that are not parts of the reactor's pressure and cooling unit and are not directly connected to the hermetic zone atmosphere

The internal configuration of the hermetic zone shall support natural circulation. Long narrow areas that support the phenomena of transition from hydrogen deflagration to detonation shall be minimised. The internal arrangement of the structures and components shall prevent buildup of explosive gases and pressure gradients in accident conditions endangering the containment or equipment inside. All separate areas where hydrogen can build up shall be fitted with an exhaust at the top. Local effects such as local hydrogen combustion (naked flame) and local detonation shall be assessed by using realistic methods. If the occurrence of such effects cannot be ruled out, the supplier shall evaluate the forces acting on the containment structure, on specific structure elements, and on the containment components.

The spatial arrangement of the containment shall be optimised so as to minimise the speed of combustion. The hermetic zone shall include large open areas. Prevention of global hydrogen detonation shall primarily rest in hydrogen dilution in the free areas. It shall be demonstrated that the design includes adequate provisions making local hydrogen detonation unlikely.

The jackets, insulation and coating of equipment and components located inside the hermetic zone shall possess the following special properties supporting their safety functions:

- Resistance to radiation up to the design dose limits
- Resistance to decontamination solutions (dilute acids, surfactants, water,...) and chemicals that are routinely used at the plant
- Resistance to physical effects
- Resistance to design basis accident conditions
- Flame does not propagate along coatings and paints
- Thermal conductivity and thickness of the layers complying with requirements for design basis accidents

Surface coatings and paints which are easy to decontaminate

When selecting the containment system components (mechanical and electrical bushings, ducts, isolation valves, packings, sealing,...), the supplier shall take into account requirements for compliance with the target tightness levels in any conditions considered within the design process. For each component, conditions shall be specified covering all the conceivable scenarios, including special operating states (e.g. leak tests) and accident situations.

The supplier shall assess, on a case-by-case basis, the potential and suitability of each components for any load in accident conditions, including their testing. The containment system components shall support the performance of the system in accident conditions.

3.6.2.2.2.1 Pressure reduction and heat removal system in the preliminary concept of the design

The system for pressure reduction and heat removal from the primary containment shall be designed in accordance with the general principles of designing safety systems as outlined in Section "3.3 Design of buildings, components, systems and equipment" and with the classification of buildings, components and systems.

The pressure reduction and heat removal system shall be equipped with secured power supply to the adequate extent. The basic requirements for secured power supply are specified in Section 3.8.1.3.3 The requirements for the backup power supply electrical systems."

The required safety functions shall be fulfilled by using combinations of different safety systems. The systems shall be based on the passive principle or on active elements or on a combination of both.

The containment design shall include the system of heat removal from the primary containment.
During design basis accidents, the system shall reduce pressure inside the primary containment by removing residual heat from the hermetic zone atmosphere and/or from water basins and releasing it into the surrounding environment.

Even during the worst design basis accident, the systems of heat removal from the primary containment shall reduce pressure to below 50% of the design pressure within 24 hours from the onset of the accident.

The system reliability shall match the design's probabilistic targets,

The system shall contain no active components inside the primary containment.

The system, as a component of the last defence-in-depth barrier, shall be adequately tight. If passive systems are used to remove heat from the primary containment, water storage shall be sufficient for 72 hours without making it up.

The system of heat removal from the primary containment intended for severe accidents shall meet basically the same requirements as specified above.

In addition, the system should be able, in design extension conditions to reduce pressure in the primary containment and stop any leaks within an adequately short period of time. The system shall be sufficiently independent of systems whose failure might contribute to reactor core damage.

The reliability of the system intended for severe accidents shall match the design's probabilistic targets.

The design shall include means for flooding the reactor shaft within a time necessary for effective heat removal from the vessel if the melt is cooled inside the vessel or for effective heat removal from the melt if the melt is cooled outside the vessel, whereby interaction of the melt with concrete is prevented.

It shall be demonstrated that criticality will not re-establish.

For the case where melt is cooled inside the reactor pressure vessel it shall be demonstrated that the load arising from the interaction between the melt and water inside the reactor vessel does not result in failure of the reactor vessel or in subsequent early failure of the primary containment (or that the probability of this is reasonably low). Furthermore, it shall be demonstrated that no fatal failure of the reactor vessel bottom resulting in a limited ability to cool the melt will occur (or that the probability of this occurrence is reasonably low). The means for flooding the reactor shaft shall be independent of the cause of the reactor core melting.

If the melt is cooled inside the vessel, the reactor shaft shall be flooded for a sufficient time before relocating the melt to the vessel bottom. For this reason, the design shall specifically take into account the requirement for reliability of the system start-up and for its operability.

For the case where the melt is cooled beyond the vessel, the design shall include a system for capturing and cooling the melt. This system shall not be equipped with active elements within the primary containment.

The system shall protect the primary containment against damage from the evolved heat, pressure of fluids or radiation to the extent that the primary containment's integrity shall not be disturbed.

It shall be demonstrated that with an acceptable probability, reactor vessel melt-through in the presence of water in the reactor shaft shall not have consequences (due to the reaction of the melt with water) that might endanger the integrity of the reactor shaft and/or the primary containment.

Long-term heat removal from the melt shall be provided.

The containment spray system, if included in the design, shall be designed primarily for post-accident reduction of the concentration of radioactive substances in the primary containment atmosphere and for pressure reduction inside the primary containment.

The design shall not consider the use of aggressive chemical additives (such as sodium hydroxide) that might cause damage to systems accommodated inside the hermetic zone or pressure reduction inside the primary containment.

For the final depressurisation of the primary containment, the design shall include the possibility of releasing non-condensable gases, evolved during a severe accident, by a controlled and monitored process which shall include filtration.

The containment design shall account for the possibility of melt ejection from the reactor at a high pressure. To counteract this, the design shall use a reliable system of depressurisation to below 2 MPa before the hazard of reactor vessel melt-through becomes imminent. Moreover, the reactor shaft or melt capturing equipment configuration shall help prevent migration of a fraction of the melt into the primary containment atmosphere. Analysis shall be performed to demonstrate that the reactor shaft and/or melt capturing equipment are capable of confining the melt and thus preventing early failure of the primary containmnet on vessel melt-through at pressures below 2 MPa.

3.6.2.2.2.2 Other facilities of the containment system in the preliminary concept of the design

The containment spray system, if included in the design, shall be designed primarily for post-accident reduction of the concentration of radioactive aerosols in the primary containment atmosphere.

The primary containment design shall include means for post-accident hydrogen concentration control in accordance with the results of analysis of hydrogen evolution during accidents.

The primary containment shall be equipped with a facility for hydrogen concentration control with catalytic recombiners and/or igniters. This facility shall cover design basis accident conditions as well as design extension conditions.

This facility, in combination with the adequately large volume of the primary containment, shall be able to keep the volume concentration of hydrogen in a dry environment within the limit of 10%, also for a 100% extent of the cladding-water reaction and actually conceivable hydrogen generation rate.

The primary containment shall have a sufficiently large internal volume to keep the mean hydrogen concentration below 13% assuming an 75% extent of the cladding-water reaction, even without taking the hydrogen concentration control system into account.

The disposition of the internals and components within the primary containment shall support efficient mixing of the primary containment atmosphere by natural circulation.

The disposition and type of the hydrogen concentration control system shall support natural convection and thus mixing of the atmosphere. All rooms in which hydrogen may accumulate shall be vented from the top. The hydrogen concentration control system shall be designed with a sufficient redundancy and margin to assure its continuous performance. The system shall be fully passive or independent of operator action for a time no shorter than 12 hours. A simple operator action may be considered for highly unlikely accidents (frequency below 10^{-7}/yr).

Hydrogen flammability limits shall not be attained even if the contribution of radiolysis to the long-term hydrogen concentration is included and release of the primary containment atmosphere into the environment is not considered.

The probability of global hydrogen detonation capable of endangering the integrity of the hermetic containment shall be sufficiently low for meeting the probability target for early (early containment system failure) and large release. If a sufficient margin to detonation is demonstrated, a short-time concentration increase to 13% in conditions of efficient steam inertisation shall be permissible. The hermetic containment design shall consider realistic combinations of other loads with the load from hydrogen recombination and global combustion (adiabatic-isochoric complete combustion) at a 10% hydrogen concentration level. The timing of the loads shall be realistic. The contribution from steam inertisation shall be specifically considered. The primary containment shall remain tight after global combustion. The geometry of the primary containment shall be optimised with respect to the combustion process development. Generally, large free areas shall be available within the primary containment. The design shall include measures to reduce the likelihood of local hydrogen detonations.

If a sufficiently low probability of local hydrogen detonation cannot be demonstrated, the effect of detonation shall be included in the design of the installation and of the primary containment.

The assessment of the long-term hydrogen concentration shall include the contribution from radiolysis and corrosion reactions. The analysis shall consider a time no shorter than 30 days from the initiation of the accident.

The evolution of other flammable gases, such as carbon monoxide, shall be analysed, and their partial pressures and combustion effects shall be considered within the primary containment design.

If the formation of noncondensable gases (such as carbon dioxide) arising from the interaction of the melt with concrete cannot be avoided, their effect shall be assessed and included in the primary containment pressure calculations. The contribution of the partial pressure of hydrogen shall be taken into account when evaluating the hermetic containment's structural integrity,

When assessing the amount of noncondensable gases, the maximum amount created by oxidation of the easy-to-oxidise materials of the reactor core shall be considered.

During normal operation, pressure inside the primary containment shall be controlled by means of the hermetic containment's ventilation system.

This ventilation system shall maintain the temperature, pressure gradients and activities inside the primary containment at levels allowing humans to enter the area.

The system shall enable increased pressure within the primary containment to be periodically relieved.

The requirements for the filtration system's capacities and capabilities shall be derived from the required activity level, expected number of inputs, and radiation protection requirements.

The design shall include online monitoring of the hermetic containment's integrity and tightness. The parameters to be monitored include temperature, humidity, gas concentrations and levels inside the hermetic envelope.

The design shall include post-accident hydrogen concentration monitoring in accident conditions. The number and layout of the sensors shall be adequate for the measurements to be representative with respect to the global hydrogen concentration as well as any local hydrogen buildup.

The operator shall have available information regarding the level inside the reactor shaft at the time preceding reactor vessel melt-through.

Furthermore, the status of emergency water reserves inside the hermetic envelope shall be available in support of accident management.

3.6.2.2.3 Passages through the containment in the preliminary concept of the design

The number of passages through the containment shall be minimised, and the passages shall meet the same requirements as put on the containment structure.

The design shall consider consequent effects, and the consequences shall be evaluated by using substantiated methods to the maximum extent, with the aim of demonstrating that no unacceptable consequent damage would occur. The load specified in Section "3.3.5 Protection against internal effects" shall be considered for the pressure parts of the components.

The design of the pipe/cable penetrations and bushings shall enable leak detection, capture and collection, and shall enable leaktightness tests against design parameters to be performed independently of the tests of the entire containment.

As a measure to minimise any radioactivity leaks into the environment, the containment shall be equipped with a isolation system enabling pipes which transport media and pass through the containment's boundary to be isolated in case of accident. It shall be demonstrated that the isolation system will also be able to perform in accident situations.

The isolation valves for automatic isolation based on an incoming isolation signal shall be designed so that they can fail in the shut position only. This requirement shall also be met in situations where their supporting systems are lost.

Mechanically driven valves with remote manual control which are not actuated by a signal for isolating the containment shall be supplied from secured power supply systems.

The isolation elements shall be equipped with leak detection systems and shall be amenable to leak tests. The outer isolation elements shall be installed as close to the containment wall as possible.

Pipe segments between the isolation valves which contain media at temperatures lower than as encountered in accident situations inside the containment shall be fitted with protection against overpressure.

The leaktight airlocks shall meet the following requirements:

- They shall be fitted with inner and outer doors controlled so that the passage remains leaktight at all times.
- Either door shall withstand the same temperatures and pressures as the hermetic envelope itself.
- The inner door shall be fitted with a press seal.

The leaktight airlocks shall be dimensioned and designed for evacuation of persons from the containment during occasional activities such as maintenance. The airlock shall be wide enough for 2 people in protective clothing carrying a stretcher with another person on it. The requirements for leaktightness of the airlocks shall be the same as those for the entire containment system.

The containment system design shall enable personnel to enter the area occasionally during normal operation. Access for planned maintenance and inspections shall be possible in any operating state, and the supplier shall provide the required facility and information regarding the operational procedures enabling this access.

The supplier shall specify measures to enable personnel to enter the area in a specified time following an accident associated with extensive contamination of the hermetic zone.

3.6.2.2.4 Tests and inspections in the preliminary concept of the design

The containment structure shall enable appropriate tests to be performed prior to starting up the installation as well as during the installation's entire lifetime. The tests shall demonstrate that the containment structure and systems meet the design and safety requirements (integrity, tightness).

The extent of the periodic containment tests required to confirm that the functional properties are met shall be specified, and design means to perform the tests shall be in place.

The requirements for the tests shall be specified in the design, and all parts that may potentially suffer damage during the tests shall be identified.

A detailed programme of containment system tests and inspections for the construction period and for the plant start-up period, complying with applicable standards, shall be developed. A controlled ageing programme shall be defined and applied during the structure's lifetime. Appropriate acceptance criteria shall be defined within that programme.

Access for testing the containment's prestressing tendons and the steel structure welds shall be provided during the construction and operational phases.

Pre-operational pressure tests shall give evidence of the integrity of all parts of the containment structure and systems. The design shall include requirements for the tests and adequate margins of the equipment to avoid damage of the structure.

The design shall enable pre-operational and period integral tests of the containment's leaktightness to be performed at the full test pressure and at lower pressures so as to enable extrapolation, The supplier shall specify the pressure for an adequate number of containment leaktightness tests during the plant's lifetime, without having to include in the design additional limitations associated with fatigue effects.

The design shall enable local individual tests of the following items to be performed during each refuelling:

- Each containment pipe penetration, in the segment between the containment isolation valves
- Mechanical and electrical penetrations
- Large passages such as installation passages or airlocks

Partial preliminary assessment

The preliminary concept of the design summarising the major requirements for the systems, structures and components and for the safety and technological functions of the safety systems described in Section "3.6.2.2 Containment system in the preliminary concept of the design" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Articles 32 to 41, IAEA SSR 2/1 [L. 252] Req. 30, 54 to 58, 5.30, 6.20 to 6.26, SÚJB Safety Guide BN-JB-1.0 [L. 276] 107 to 117, and WENRA [L. 27] App. E 4, 9.8, K 3.13.

3.6.2.3 EMERGENCY CORE COOLING SYSTEM IN THE PRELIMINARY CONCEPT OF THE DESIGN

This part of the Initial Safety Analysis Report summarises basic design requirements for selected systems of the emergency core cooling system. At the next stage of the licensing documentation, this section shall be completed with additional safety systems at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.6.2.3.1 Purpose and description of the system in the preliminary concept of the design

The purpose of the emergency core cooling system is to prevent the development of design basis accidents into severe accidents associated with the hazard of loss of integrity of the containment system.

The design shall include an emergency core cooling system performing the following functions during design basis accidents:

- Heat removal from the reactor core for an adequately long time, i.e. making up and cooling the reactor core, controlled depressurisation of the primary circuit, and long-term residual heat removal
- Reactivity control, i.e. addition of boric acid to the primary coolant
- Confinement of radioactive materials (i.e. reactor core cooling to avoid fuel damage bringing about radionuclide leak beyond the primary circuit) and maintaining the same integrity level as required for the containment system

The next sections describe the safety system(s) meeting the above goals / performing the above functions.

3.6.2.3.2 Principles of the solution in the preliminary concept of the design

The design of the emergency core cooling system shall comply with the general design principles for safety systems specified in Section "3.3 Design of buildings, components, systems and equipment" in accordance with the classification of buildings, components and systems.

The emergency core cooling system shall be adequately supplied from a secured power supply system. The basic requirements for secured power supply systems are specified in "Section 3.8.1.3.3.

The required safety functions shall be fulfilled by using combinations of different safety systems. The systems shall be based on the passive principle or on active elements or on a combination of both.

3.6.2.3.3 The emergency core cooling system in the preliminary concept of the design

The design shall include an active or passive system of emergency primary circuit coolant makeup.

This system shall provide an emergency core cooling system during a design basis accident, such as loss of primary coolant accident, mean steam-line break accident, or primary coolant level decrease accident during fuel reloading. By supplying an adequate amount of coolant, this system shall reduce the extent of fuel damage, thereby reducing any subsequent radionuclide leaks to below the acceptability limits specified for design basis accidents.

Active safety injection system

The active safety injection system shall contain:

- A subsystem for emergency primary coolant makeup with suction from the in-containment refuelling water storage tank, located inside the primary containment
- A subsystem for passive makeup by means of hydro accumulators
- Heat exchanger for heat transfer to the inserted cooling circuit

The active safety injection system, in combination with the depressurisation system, shall perform the following functions:

- Primary coolant makeup to secure a sufficient amount of coolant in the reactor vessel so that the reactor core is adequately cooled in the event of a loss-of-coolant accident (LOCA) (the system will provide makeup if the leak exceeds the capacities of the normal makeup system)
- High-strength pressure boundary of the same protection level as the primary containment (if a part of the system is located beyond the primary containment)
- Residual heat removal after a large break or intermediate break loss-of-coolant accident (LB LOCA or IB LOCA)
- Supply of boric acid in an amount required for a safe shutdown (this diverse reactivity control function may also be fulfilled by an alternative dedicated system)
- Supply of coolant with boric acid solution to make up for coolant contraction and to reduce reactivity in the event of steam-line break, up to the size of a two-sided outflow from the main steam-line.
- Prevention of an excessive boric acid concentration following cold leg LOCA

- Coolant makeup in the event of a steam generator tube rupture (SGTR) accident, with appropriate restriction of primary containment bypass.
- Residual heat removal by the feed and bleed method in design extension conditions where heat removal through the steam generators is unavailable
- Coolant makeup in the event of unwanted level decrease in the reactor unit outage regime (aftercooling and refuelling, i.e. operation at a level corresponding to the level of the loops)

The primary coolant makeup system shall provide an amount of coolant sufficient to prevent any fuel damage for breaks up to an equivalent diameter of 150 mm.

The delivery of the emergency core cooling system shall be set to a level below that for the opening of the relief valves in the secondary circuit (on the steam-lines). If this level cannot be complied with, it shall be demonstrated that no primary containment bypass will take place.

The system delivery pressure shall be set to a level below the primary coolant circuit pressure during normal operation in order to avoid makeup in the event of inadvertent start-up of the system.

Adequate pressure at the pump suction shall be provided in all circumstances. The suction pipe shall be directed downwards from the containment sump (with no knees, etc.)

The active safety injection system shall provide adequate cooling of the makeup water so that the amount of steam leaking into the primary containment in the event of large-break or intermediate-break loss-of-coolant accident is reduced rapidly.

The makeup water shall be cooled via an intermediate water system.

Analysis shall be performed to demonstrate that the pressure at the pump suction will be adequate during any primary pipe break accident. This analysis shall consider the temperature inside the primary containment, pressure losses at the sieves of the pits, pump delivery, and uncertainties of the pressure losses inside the suction piping.

The pool for water for fuel reloading shall serve as the source of boric acid.

Alternative boric acid solution sources may become available during some accidents.

The concentration of the stored boric acid solution shall not exceed 7000 ppm (approx. 40 g/L).

The system shall be used as a safety means for boric acid addition during design basis accidents. The system shall not be automatically activated during abnormal operation.

Boric acid concentration increase by using the emergency core cooling system shall be achieved in combination with the primary circuit depressurisation system.

The active safety injection system of the primary circuit shall provide boric acid addition in case of main steam-line break.

The amount of boric acid added by the active safety injection system shall be adequate to keep the reactor in a safe shutdown state at a coolant temperature equal to ambient temperature also if the control cluster with the highest weight remains fully withdrawn from the reactor core.

The makeup and heat removal provided by the active safety injection system shall be adequate to attain conditions required for the feed and bleed regime, viz. by means of the pressuriser relief valves or by means of the safety system of primary circuit depressurisation.

The storage pool for water for fuel reloading should be located within the hermetic zone and:

- provide a higher protection against external effects
- manage steam removal during primary circuit depressurisation
- provide an internal water margin for managing severe accidents

This pool shall store water with boric acid for the spent fuel pool, fuel reloading shaft, active safety injection system, and containment spray system (if planned in the design); if located within the primary containment, it can also be used for condensation of steam from the pressuriser.

Suction for the active safety injection system shall be set during any state.

The suction shall be protected against clogging by means of redundancy, physical separation, and installation of an anti-clogging system.

Passive safety injection system

The extent of emergency coolant makeup and emergency primary circuit cooling provided by the passive safety injection system shall be adequate to reduce fuel damage and subsequent radionuclide leaks.

The system shall include equipment for reactor core flooding and for adequate heat removal by means of coolant tanks for making up at low pressures (in-containment refuelling water storage tank), coolant tanks for making up at low pressures (hydro accumulators and core make-up tanks).

The passive safety injection system, in combination with the depressurisation system, shall perform the following functions:

- Make up primary coolant during design basis accidents or in design extension conditions in situations where normal make-up is no longer sufficient
- Make up primary coolant during loss-of-coolant accidents
- Add boric acid to the primary coolant during design basis accidents or in design extension conditions if normal make-up is no longer sufficient
- Add boric acid to the primary coolant so as to attain and maintain a safe shutdown and to reduce reactivity increase after main steam-line break
- Prevent the presence of a too high boric acid concentration in the primary circuit following a large-break loss-of-coolant accident (LB LOCA) by providing adequate flow through the reactor core

The passive safety injection system shall enable pH control following a loss-of-coolant accident in order to reduce stress corrosion and increase iodine capture.

In addition, the system shall provide a margin for maintaining water level in the reactor shaft and for cooling the spent fuel storage pool during refuelling.

The system shall make up fuel to an extent adequate to comply with the fuel limits for design basis accidents.

The system shall provide an amount of coolant sufficient to prevent any fuel damage for breaks up to an equivalent diameter of 150 mm.

Boric acid shall be stored in core make-up tanks or in hydro accumulators.

The concentration of the stored boric acid solution shall not exceed 7000 ppm (approx. 40 g/L).

The amount of boric acid added by the passive safety injection system shall be adequate to keep the reactor in a safe shutdown state also if the control cluster with the highest weight remains fully withdrawn from the reactor core.

Safe shutdown with the use of a passive safety injection system corresponds to a coolant temperature of 100°C.

The passive safety injection system shall ensure subcriticality after a main steam-line break accident.

From the long-term aspect, the passive safety injection system, by using the core make-up tanks or hydro accumulators, shall ensure subcriticality of the reactor also after the temperature has decreased to the ambient level.

The passive safety injection system shall control boric acid concentration automatically, without operator action, so as to prevent borate deposition in the reactor core following a cold leg LOCA.

After a cold leg LOCA, the system shall prevent borate deposition by means of coolant make-up (washing out).

The passive safety injection system shall comprise a set of equipment, each item having its own suction and make-up piping.

No additional setting/adjustment shall be necessary once the system has been activated.

All passive safety injection system components shall be accommodated in the primary containment.

The number of hydro accumulators shall be reduced to the minimum required for performing the safety function.

The size of the hydro accumulators and the make-up speed shall be adequate to enable the reactor vessel to be flooded in the event of a loss-of-coolant accident.

In addition, in the event of a small-break LOCA, the hydro accumulators together with the core make-up tanks shall keep the amount of coolant large enough to reach above the reactor core level, until makeup from the in-containment refuelling water storage tank is available.

The water reservoir for the passive safety injection system (in-containment refuelling water storage tank) shall contain:

- Water storage for fuel reloading
- Water for primary coolant circuit make-up by gravity filling
- Heat sink for passive heat exchange and for condensation of steam from the pressuriser

The volume shall be adequate to flood the refuelling shaft during fuel reloading and, at the same time, to support the operation of the passive safety injection system.

The suction shall be protected against clogging by means of redundancy, physical separation, and installation of an anti-clogging system.

The size of the core make-up tanks shall be adequate to:

- keep, together with the hydro accumulators, the amount of coolant large enough to reach above the reactor core level after a small-break LOCA, until makeup from the in-containment refuelling water storage tank is available
- provide a sufficient amount of coolant to the reactor core after a large-break LOCA, until make-up from the in-containment refuelling water storage tank is available
- minimise the probability of inadvertent start-up of the safety depressurisation system (if included in the design)

Safety depressurisation system

The safety depressurisation system shall perform the following functions:

- Primary circuit depressurisation if the pressuriser spray system is unavailable in regimes of cold shutdown with natural circulation
- Enable the primary circuit to be depressurised for feed and bleed and for emergency make-up during some design basis accidents (e.g. IB LOCA) or in design extension conditions (e.g. total loss of make-up water)
- Enable the primary circuit to be depressurised with a view to avoiding core melting at a high pressure
- Enable the primary circuit to be depressurised after steam generator tube rupture
- Enable the primary circuit to be depressurised for boric acid addition following exceedingly heavy cooling of the steam generator secondary side
- Enable emergency removal of non-condensable gases from below the reactor vessel head and other high-lying points of the primary circuit

The depressurisation system reliability shall match the design's probabilistic targets.

The system shall minimise opportunities for inadvertent or wrong start-up.

If the system is started up inadvertently, the reactor shaft shall not be flooded up to the level of the reactor vessel bottom.

3.6.2.3.4 In-service inspection of the emergency core cooling system in the preliminary concept of the design

The emergency core cooling system shall be designed so as to enable its in-service calibration, testing, maintenance, repair, and inspection or monitoring with respect to its performance qualification, for its entire service lifetime. If some parts of the system are inaccessible for such in-service activities, safety measures to make for any undetected failures shall be in place.

3.6.2.3.5 Residual heat removal system in the preliminary concept of the design

Residual heat shall be removed by an active or passive residual heat removal system which will provide long-term residual heat removal from the reactor core to the ultimate heat sink (air) during design basis accidents. Residual heat removal from the core helps reduce fuel damage and, thereby, any radionuclide leaks, to within the acceptability criteria for design basis accidents.

Active residual heat removal system

The active residual heat removal system shall include a system of heat removal through steam generators (see the emergency steam generator supply system below) and a system of heat removal in conditions where heat removal through steam generators is inefficient or unavailable (see the active residual heat removal system below).

The emergency feedwater system shall:

- supply feedwater to the steam generators in order to remove heat from the primary circuit until attaining conditions allowing the active residual heat removal system to be connected (in conditions where both the main and standby steam generator supply systems are unavailable)
- remove heat from the primary circuit during a small break LOCA with circulation preserved, and contribute to heat removal by steam condensation in steam generators during an intermediate break LOCA with loss of circulation in the primary circuit

The system's design parameters (minimum flow rate, number of steam generators fed, amount of water stored,...) shall be set so that conditions allowing the active residual heat removal system to be connected and to perform are attained during any design basis accident, without having to make up water to the emergency feedwater supply system storage tanks.

Feedwater storage shall be adequate for attaining conditions allowing the active residual heat removal system to be connected and to perform.

Design basis accidents (such as main feedwater pipe break with automatic isolation of the main steam-line and isolation of the damaged steam generator in 30 minutes) shall not bring about bulk boiling or reactor core exposure.

The system shall serve as a standby system for the standby feedwater system during anticipated operational occurrences, e.g. during blackout or during main feedwater system loss. In such situations the system shall:

- prevent complete pressuriser flooding with water and ingress of water into the pressuriser's safety valves
- prevent uncontrolled leak from the bubble condenser tank to the primary containment

Adequate redundancy shall be provided, i.e. maintenance of one leg of the system, sufficient to avoid the necessity to pass to the feed and bleed regime, shall be assumed during any design basis accident or in any design extension conditions.

Feedwater supply shall be adequate for attaining and maintaining the reactor in the safe shutdown state considering:

- refilling of the leaktight steam generators
- 8-hour operation in the hot zero power mode
- subsequent primary circuit cooling within 6 hours to attain conditions allowing the active residual heat removal system to be operated

- continuous operation of 1 main coolant pump

The feedwater storage shall be adequate for:

- operation in the hot shutdown mode for 24 hours or longer, with subsequent transition to a state allowing the active residual heat removal system to run in conditions of natural circulation in the primary circuit
- operation in the hot shutdown mode, with subsequent transition to a state allowing the active residual heat removal system to run during station black-out, taking into account non-safety-related feedwater sources.

The safety feedwater storage shall be adequate to allow cooling in conditions of natural circulation in the primary circuit.

It shall be possible to make up the feedwater storage by using the non-safety-related system.

The emergency feedwater system shall be equipped with a minimum of 2 tanks, either being capable of providing 50% of the required amount of feedwater.

The total capacity of all tanks shall support operation in the hot shutdown state with natural circulation in the primary circuit for 24 hours.

The design shall include equipment for making the tanks up with water from other systems even in station black-out situations.

It shall be possible to interconnect the tanks.

The active residual heat removal system shall cool the primary circuit adequately in situations where heat removal through the steam generators is inefficient or unavailable. In regimes with the reactor shut down, including fuel reloading, heat shall be transferred continuously from the primary circuit to an intermediate cooling circuit. This system may be combined with the safety injection system.

The active residual heat removal system shall consist of:

- pumps for primary coolant circulation from the primary circuit to the heat exchangers and, after cooling, back to the primary circuit
- heat exchangers for heat transfer to the intermediate cooling circuit

The design shall include a minimum of 2 branches, each with its own suction from the primary circuit and its own delivery to the primary circuit.

The active residual heat removal system shall be designed to:

- remove heat from the primary circuit during normal or emergency cooldown after achieving partial cooldown by steam generators
- remove heat from the primary circuit after attaining normal or emergency cold shutdown
- keep the desired temperature in the primary circuit during fuel reloading, maintenance, or other work on the shut-down reactor
- provide adequate flow through the reactor core in regimes with the main coolant pumps inoperable
- provide a high-strength barrier of the same level as that of the containment

- remove heat from the primary circuit in safe shutdown regimes after attaining partial cooldown by other means, in particular, long-term core cooling in accident situations such as main steam-line break or small-break LOCA
- provide protection against cold primary circuit overpressure

The active residual heat removal system shall be located within the containment system (including the secondary containment).

Should any part of the system be located beyond the primary containment, it must be demonstrated that the hazard of containment system bypass is minimised by means of reliable isolation elements.

The active residual heat removal system shall be equipped with means for protection against overpressure. Those means shall be located in the primary containment.

The active residual heat removal system shall be designed to reduce any leaks in conditions of thermal overpressurising of the system.

The active residual heat removal system shall preferably be routed continuously downwards to the system's pumps.

The active residual heat removal system shall be designed to provide adequate pressure for pump suction in any circumstances.

No risk of pump suction loss in a regime with the liquid level lowered down to the level of the loops shall exist.

Passive residual heat removal system

The passive residual heat removal system shall:

- remove heat from the primary circuit in situations where heat removal through the steam generators by means of the active systems is inefficient or unavailable
- remove heat from the primary circuit in situations where the process heat removal system is unavailable

The passive residual heat removal system shall comprise heat exchanger(s) for heat transfer by natural circulation as far as the ultimate heat sink.

It shall be designed for the primary circuit design pressure and temperature.

It shall possess a capacity adequate to prevent opening of the pressuriser safety valve during accidents with total loss of main steam generator feedwater.

It shall enable the cooling rate to be controlled.

The passive residual heat removal system shall be designed to enable natural circulation between the system and the primary circuit.

It shall provide cooling also in conditions of the maximum design parameters of the primary circuit and the primary containment even during the worst single failure and without using alternating currect supply.

The passive residual heat removal system's heat sink shall be adequate for operation for 72 hours, and the possibility of long-term make-up shall exist.

3.6.2.3.6 Fire protection in the preliminary concept of the design

The safety injection system and the residual heat removal system shall be designed so as to minimise the frequency and impacts of fires on the safety function of those systems and enable the condition of the nuclear installation during and after fire to be monitored (see Section "3.9.2.5.1 Fire protection programme").

Partial preliminary assessment

The preliminary concept of the design summarising the most important requirements for the emergency core cooling system outlined in Section "3.6.2.3 Emergency core cooling system in the preliminary concept of the design" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Articles 22(1, 2, 6), 25(1, 2), 26 (a, b), and 27, IAEA SSR 2/1 [L. 252] Req. 47, Req. 48, Req. 51, Req. 52, Req. 53, 6.13 to 6.16, 6.18, 6.19; SÚJB Safety Guide BN-JB-1.0 [L. 276] (56, 57, 71, 75-79, 81, 116, and 118); and WENRA [L. 27] App. E 9.7, Issue 2.1.

3.6.2.4 HABITABILITY SYSTEMS OF THE CONTROL ROOMS IN THE PRELIMINARY CONCEPT OF THE DESIGN

This section summarises the preliminary concept of the design and the requirements for the design of habitability systems of the control rooms. In the Preliminary Safety Analysis Report that will follow, this Section will contain description of the design basis for the safety-relevant components, description of the system, safety assessment, requirements for inspections, tests, and instrumentation, control, blockages, and signalizations.

3.6.2.4.1 Purpose and description of the system in the preliminary concept of the design

The habitability systems of the control rooms shall provide such parameters of the environment (air temperature, humidity, cleanliness, flow velocity; noise pressure level; pressure with respect to the surround areas; light intensity; ...) in the control rooms (both main and supplementary control room) which provide adequate comfort, safety and health conditions for personnel and maintain the installed I equipment appropriately operable in any operating state as well as in accident conditions. Internal as well as external effects outlined in Section "3.3.3 Protection against external effects" shall be taken into account in the design.

3.6.2.4.2 Operating control room in the preliminary concept of the design

In normal operating conditions, the following requirements of the habitability system of main control room shal be provided:

- continuous supply of fresh air
- Maintaining relative air humidity acceptable for both the personnel and equipments
- removal of heat dissipated by the operating equipment and other equipment, personnel, and also by external heat sources or heat transferred from the adjacent rooms, so that the maximum permissible temperature for indoor areas is not exceeded
- heating to ensure the minimum permissible ambient temperature.

In accident conditions, the the following requirements of the habitability system of main control room shall be provided:

- continuous supply of fresh air
- removal of heat dissipated by the operating equipment and other equipment, personnel, and also by external heat sources or heat transferred from the adjacent rooms, so that the maximum permissible temperature for indoor areas is not exceeded

In all operating states as well as in accident conditions, a slight air overpressure shall be maintained in the main control room in order to prevent polluted air (smoke, explosive or toxic gases, radioactivity) from penetrating into the room.

Noise limits stipulated by applicable rules and regulations shall be complied with in the control room.

The fresh air inlet system shall be equipped with adequate filters.

3.6.2.4.3 Supplementary **control room in the preliminary concept of the design**

The supplementary control room will be used if the main control room is not available. It shall be physically and electrically separated from the main control room.

The habitability systems of the supplementary control room shall provide the operators with adequate comfort, safety and health conditions, and shall maintain parameters of the environment suitable for the installed technological equipment to remain operable in any operating state as well as in accident situations. The probabilistic approach may be used to assess a situation when an accident has occurred and, at the same time, the main control room is not available, as a basis for specification of the conditions of habitability of the supplementary control room in case of accident.

Partial preliminary assessment

The preliminary concept of the design summarising the key requirements for the systems to maintain the control rooms inhabitable, as described in the Section "3.6.2.4 Systems for maintaining the control rooms inhabitable" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 20(2, 3, 4); IAEA SSR 2/1 [L. 252] Req. 65, 6.39, 6.40; Req. 66, 6.41; and SÚJB Safety Guide BN-JB-1.0 [L. 276] (94, 96).

3.6.2.5 SYSTEMS TO CONTROL ACCIDENT LEAKS OF RADIOACTIVE MATERIALS IN THE PRELIMINARY CONCEPT OF THE DESIGN

This section summarises the preliminary concept of the design and the requirements for the design of the systems to control leaks of radioactive materials in accident situations. At the next stage of the licensing documentation, this section shall be completed with additional information at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.6.2.5.1 Purpose and description of the system in the preliminary concept of the design

The safety systems to reduce (control) radioactivity leaks in accident situations (such as ventilation systems with air filtration, spray system) shall reliably reduce the release and/or migration of radioactive substances in the environment to a level As Low As Reasonably Achievable (ALARA), which in design basis accident situations shall be lower than the regulatory limits. Their safety function rests in reducing the volume activity and/or modifying the composition of the radionuclides in accordance with the acceptance criteria.

3.6.2.5.2 Principles of the solution in the preliminary concept of the design

The systems to control leaks of radioactive materials in accident situations shall be designed in accordance with the general principles for designing safety systems, as outlined in Section "3.3 Design of buildings, components, systems, and equipment" and with the classification of buildings, components and systems.

The systems shall be equipped with secured power supply to an extent as appropriate. The basic requirements for backup power supply are specified in Section 3.8.1.3.3. The requirements for the backup power supply electrical systems."

The required safety functions shall be fulfilled by using combinations of different safety systems. The systems shall be based on the passive principle or on active elements or on a combination of both.

The provisions to control radioactivity leaks shall primarily include capture of the maximum feasible amount of the fission products within the primary containment.

Measures to control radioactivity leaks from the primary containment shall include the natural processes of washing-out, deposition and capture (dying away) of the radionuclides and, as appropriate, the use of active systems supporting such processes.

In order to better capture radioiodine, the pH in the pools within the primary containment shall be kept slightly basic after accidents.

The containment spray system, if included in the design, shall be designed primarily for post-accident reduction of the concentration of radioactive aerosols in the containment atmosphere.
No aggressive chemical additives (such as sodium hydroxide) that might cause damage to the systems accommodated within the primary containment shall be used.

For the final depressurisation of the primary containment, the design shall include the possibility of releasing non-condensable gases, evolved during a severe accident, by a controlled and monitored process which shall include filtration.

The design shall include the secondary containment which will additionally reduce radioactivity leaks from the primary containment. For this purpose it shall encompass the highest feasible number of hermetic containment penetrations and passages.

The primary containment shall be tight enough to reduce the accident doses down to the regulatory limits. Reduction in the amount of the radioactivity leaked shall be achieved through natural processes, such as deposition of aerosols and capture (dying-out) of noble gases, or by using active emergency ventilation systems.

The possibility of bypassing the containment shall be minimised.

During normal operation, the containment's internal space shall be held at a reduced pressure against the surroundings constantly by means of the ventilation system, and the discharges shall be filtered as appropriate.

If the design includes an emergency containment ventilation system, this system shall:

- be independent of the operational ventilation system as much as possible
- keep reduced pressure in the containment's internal areas
- collect and (as appropriate) filter leaks from the containment

The emergency ventilation system filters shall be resistant to the heat that may be produced by the radioactive decay of the radionuclides trapped.

3.6.2.5.3 Fire protection in the preliminary concept of the design

The requirements for fire protection of buildings and systems containing radioactive materials are specified in Section "3.9.1.5.1 Fire protection programme".

Partial preliminary assessment

The preliminary concept of the design summarising the key requirements for the systems to control radioactivity leaks, as described in Section "3.6.2.5.1 Systems to control leaks of radioactive substances in the preliminary concept of the design" creates preconditions for compliance with the requirements stipulated by Decree No. 195/1999 Coll. [L. 266] Articles 42(1), and 44; IAEA SSR 2/1 [L. 252] Req. 55, 58, 81, 6.29, 6.71, 6.72; and SÚJB Safety Guide BN-JB-1.0 [L. 276] 116, 119.

3.6.3 COMPREHENSIVE PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT

The ETE3,4 design shall meet the safety requirements specified in Section 3.3.1.1 as well as the requirements specified in Sections 3.6.1.1 to 3.6.1.5, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and related implementing decrees covering issues of nuclear safety, radiation protection and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270].

To meet those requirements, the ETE3,4 design shall use systems, structures and components and their equipment integrated within safety systems such as shall ensure (with adequate reliability and resistance) qualitatively and quantitatively adequate safety and technological functions in accordance with the specified requirements for the prescribed safety functions.

The principles of the design solution described in Section "3.6.2 Description of the design for the purpose of the preliminary assessment" were set up based on the requirements that the applicant for licence imposed on the potential suppliers of the nuclear installation within the tender, and they make up the concept of the design solution for this part of the design. The partial assessments give evidence that the expected design of the safety systems creates preconditions for compliance with the relevant requirements for systems, structures, and components and safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], SÚJB BN-JB-1.0 Safety Guide [L. 276], and the IAEA SSR 2/1 [L. 252] and WENRA [L. 27] documents.

The particular method of implementation of the individual requirements for the performance of the safety systems described in Section 3.6.2 shall be specified in detail in the nuclear power plant design.

3.7 INSTRUMENTATION AND CONTROL SYSTEMS

This comprehensive part of the Initial Safety Analysis Report is structured into three basic sections:

Opening Section 3.7.1 summarizes, analyses, and specifies the basic legislative requirements for instrumentation and control systems, including the general requirements for their structure, the requirements for computer based system applications in systems important to safety, the requirements for protection systems, safe reactor shutdown and information systems important to safety. The section also includes the requirements for systems that are not required for nuclear safety and data communication systems.

Section 3.7.2, which follows, contains the description and specification of the basic requirements for the instrumentation and control system functions specified in the opening Section 3.7.1 in a form of “envelope” applied design requirements for relevant subsystems of the instrumentation and control system within all the relevant designs involved in the current tender procedure. The section's goal is to formulate the design's general properties for the purposes of partial preliminary assessment.

The final section 3.7.3 contains a comprehensive preliminary assessment of the concept of the instrumentation and control systems design that summarizes the conclusions of the partial preliminary assessments completed in section 3.7.2. The assessment of the summarized requirements contains a tentative design concept assessment required by the law. The applicant shall use the next section of the licence documentation in order to provide assessment and supporting information on the selected design that will allow to assess the instrumentation and control systems' ability to fulfill the specified safety functions during the nuclear unit's whole lifetime under any operating conditions, including accident conditions. The section shall also be supplemented with more detailed information featuring at the depth and in the structure described in RG 1.206 [L. 275].

3.7.1 THE BASIC LEGISLATIVE REQUIREMENTS FOR THE INSTRUMENTATION AND CONTROL SYSTEMS

Instrumentation and control systems provide monitoring and control functions of technology systems supporting power production at a nuclear power plant, especially nuclear systems important to safety. The main task of the instrumentation and control systems is to maintain the specified operating parameters of technological systems in line with the design criteria, limitations, and safe operation conditions, and in the case of systems important to nuclear safety, it will also enable reliable fulfillment of all required safety functions.

Note:
The requirements for the radiation monitoring systems are specified in Section "3.11.1.5 ***Operating media and discharge radiation monitoring****" in accordance with the agreed section structure 3 ZBZ.*

3.7.1.1 THE GENERAL REQUIREMENTS FOR THE INSTRUMENTATION AND CONTROL SYSTEMS

The section contains the general requirements for all the instrumentation and control systems in relation to all the instrumentation and control subsystems. The section elaborates more detailed information on the general safety requirements and applies them to the instrumentation and control systems.

The instrumentation and control systems shall adequately meet the basic requirements specified in section "3.3.1.1 The basic safety requirements".

Reference: SÚJB BN-JB-1.0 Safety Guide (20), IAEA SSR 2/1 Req. 9

The instrumentation and control systems shall meet the requirements of the basic conceptual Czech national standards or the supplier's equivalent norms and also the Czech legislature's requirements relating to employee safety and health protection. The utilized technical regulations, norms, requirements, rules, computation programs used for instrumentation and control system designing shall be clearly specified and their suitability for nuclear facilities shall be secured and verified and they shall be in line with the internationally accepted practice. If their combination is used, it shall form a consistent whole, and the mutual compatibility and usability of its individual components shall be proven. Foreign documents and programmes must guarantee such a level of justified interest protection that is identical to the one in the Czech Republic.

In conformity with section 3.3.1.1, there are also the requirements deriving from the application of these basic safety requirements, specifically to the instrumentation and control systems, i.e. the control, information, and protection systems:

- Defence in depth
- Quality requirements
- Classification
- Equipment failure protection
- The requirements for the control systems important to nuclear safety
- The requirements for the separation of control and protection systems

These general requirements apply to all the instrumentation and control systems. The level of their application depends on a specific system's importance in terms of nuclear safety (i.e. on its safety classification).

Requirements for defence in depth

Reference: Decree No. 195/1999 Coll., Article 3, SÚJB BN-JB-1.0 Safety Guide (5, 10, 12, 21), IAEA SSR 2/1 Req. 7, 4.9, 4.11

The defence in depth principle shall also be prepared and applied to the instrumentation and control systems in order to limit any impacts of instrumentation and control system failures negatively affecting the monitoring, control and protection functions of systems important to nuclear safety. The current application of this procedure results in such an architecture of instrumentation and control systems that shall ensure the application of suitable protective measures at a few independent levels; therefore, any deviation or failure shall be detected and compensated for via adequate countermeasures. In the case of a lower level measure failure, higher level measures get applied during the next step.

References: SÚJB BN-JB-1.0 Safety Guide (11), IAEA SSR 2/1, Req. 7, 4.10

During the design, measures that generally prevent the power plant's operation upon one barrier failure despite the existence of multiple barriers shall be implemented. All the protection levels shall be permanently available and any deviations from this rule shall be sufficiently justified and the sufficient meeting of the prevention and effect mitigation principle shall be proven. During design basic accidents, a sufficient number of barriers shall remain preserved in order to meet the general safety goal.

Quality requirements

Reference: Decree No. 195/1999 Coll., Article 4 (1)

The instrumentation and control systems that are important for the nuclear power plant's nuclear safety and radiation protection shall secure their reliable operation during normal and abnormal operation and also under accident conditions, including the ability to mitigate the impacts of failures and accidents, as specified in the design.

References: Decree No. 195/1999 Coll., Article 4 (2), SÚJB BN-JB-1.0 Safety Guide (41, 42), IAEA SSR 2/1 Req. 24, 25

The durability and robustness of the instrumentation and control systems architecture and the level of reliability, redundancy, diversity and independence of the systems shall be adequate to the requirements for robustness, reliability, and other functional requirements that generally derive from the requirements of the monitored and controlled ETE3,4 technological systems (especially the mechanical-technical, construction, and electrical systems).

The technical solution design of the instrumentation and control systems shall feature safety measures against:

- Single failures
- Common cause failures

References: Decree No. 195/1999 Coll., Article 4 (2), IAEA SSR 2/1 Req. 29

The nuclear safety instrumentation and control systems design shall enable to check and test their functional capabilities and reliability, complete inspection and repair activities and calibrations without any negative impacts on nuclear safety under all the conditions specified in the design.

References: SÚJB BN-JB-1.0 Safety Guide (85)

The requirements for the uniformity and correctness of measuring and measuring devices are specified in a special legal regulation (Act No. 505/1990 Coll., on metrology, as amended).

References: SÚJB BN-JB-1.0 Safety Guide (19), IAEA SSR 2/1 Req. 9, 4.14

The design concerning the instrumentation and control systems that are important to nuclear safety shall prefer the application of such systems, componets and design methods that are in line with the relating regulations and specified norms and that have been tested through equivalent applications. The contribution of new methods and technologies must be proven and supported through sufficient experiences, tests, or research.

References: IAEA SSR 2/1 Req. 29

The design of instrumentation and control systems important to safety shall guarantee their ability to complete the given tasks under all the conditions specified within design basis during all of their life cycle phases.

Classification requirements

Reference: Decree No. 195/1999 Coll., § 4 (1), SÚJB BN-JB-1.0 Safety Guide (6, 7), IAEA SSR 2/1 Req. 22, 23, WENRA App. E 10.2

The instrumentation and control systems shall meet the requirements specified in the ETE3,4 design concerning monitoring, measuring, and recording functions and the control of technology systems operating parameters specified in the machine-technology, construction, and electrical section. These requirements for the instrumentation and control systems will be based on the basic classification of systems, structures and components on important to nuclear safety and on not required to nuclear safety, and from the categorization of selected types of systems, structures and components into safety classes (BT) based on safety function significance that the the given system (structure, component) performs.

Method of classification safety function significance shall be based on a deterministic approach supplemented with a probability approach while considering these factors:

- impacts of the safety function's failure
- frequency of the required safety function application
- how long after a postulated initiation event and how long after initiation is the system (structure, component) supposed to act

The safety class classification criteria for the selected equipments are specified in a special decree (Decree No. 132/2008 Coll. [L. 258]).

This safety classification shall form the basis for the determination of specific requirements for the design solution's robustness, the qualification and reliability of the instrumentation and control systems, including the requirements for these systems' parameters (speed of signal processing, the accuracy of measuring and control circuits, time response to input event, and other parameters).

Equipment failure protection

Reference: Decree No. 195/1999 Coll., Article 7, SÚJB BN-JB-1.0 Safety Guide (43), IAEA SSR 2/1 Req. 30, 5.48

The design of instrumentation and control systems shall specify the equipments properties and measures that shall make sure the equipments is able (qualified) to cope with the ambient environmental conditions (for example, temperature, humidity, radiation) that it may be exposed to while supporting the nuclear facility, as required by its design function, while considering the equipments' degradation due to the expected conditions created by the surrounding environment.

The instrumentation and control systems and their components shall also be designed to cope with the surrounding environment's design properties, including EMC.

References: IAEA SSR 2/1 Req. 26

The design solution for systems and components shall be conveniently based on the fail-safe design.

The requirements for the control systems important to nuclear safety

Reference: Decree No. 195/1999 Coll., Article 16 (1), SÚJB BN-JB-1.0 Safety Guide (83, 95), IAEA SSR 2/1 Req. 59, 60, WENRA App. E 10.1

The nuclear power plant shall be equipped with control systems allowing to monitor, measure, record and control the operating parameters essential for securing nuclear safety during normal and abnormal operation and under accident conditions. These systems shall secure the maintaining of the operating parameters of equipments important to nuclear safety in line with the design criteria, limitations, and safe operating conditions.

The control and information systems shall provide the necessary visual and audio warnings informing of new or changed operating states and processes that deviate from the admissible limits for the normal operation and may affect nuclear safety.

References: IAEA SSR 2/1 Req. 62

The design of instrumentation and control systems important to nuclear safety shall consider the requirements for their high reliability and the requirements for periodic testing adequate to performed safety functions.

The requirements for the separation of control and protection systems

Reference: Decree No. 195/1999 Coll. Article 19 (1), SÚJB BN-JB-1.0 Safety Guide (88), IAEA SSR 2/1 Req. 22, 64

The protection and control systems will be separated so that a fault of the control systems does not affect the capability of the protection systems to perform the required safety function. The functionally necessary and practical connection of the protection and control systems will be limited as much as possible so that it does not have a significant effect on nuclear safety.

References: IAEA SSR 2/1 Req. 64, 6.38

If signals are used in common by both a protection and any control systems, their separation shall be ensured, and those signals shall be classified as part of the protection system.

3.7.1.1.1 The requirements for the use of computer systems in systems important to safety

References: SÚJB BN-JB-1.0 Safety Guide (92), IAEA SSR 2/1 Req. 63, 6.37, WENRA App. E 10.10

If computer based equipments are included in the protection systems design, an adequate certificate of quality, including its independent assessment, shall be required. If it is not possible to prove the reliable completion of designed safety functions of the protection systems sufficiently enough, the protection system's function shall be backed up on a different functional basis in order to avoid common cause failures as much as possible.

References: IAEA SSR 2/1 Req. 39

Unauthorized access to or tampering with the systems important to nuclear safety and computer hardware and software shall be prevented.

References: Decree No. 195/1999 Coll., Article 7, 19, SÚJB BN-JB-1.0 Safety Guide (42, 88), IAEA SSR 2/1 Req. 22, 23, 39

If the instrumentation and control systems important to nuclear safety are implemented on computer based technology, the design shall feature adequate codes and standards necessary to secure sufficient level of cyber security.

References: IAEA SSR 2/1 Req. 63, 6.37, WENRA App. E 10.10

If the nuclear power plant's instrumentation and control systems important to nuclear safety are computer based, the design shall feature adequate codes and standards necessary to develop and test computer hardware and software for the whole life cycle of computer based systems and that shall especially apply to the whole software development cycle. The complete computer based systems development shall be subject to an adequate quality assurance programme.

The design shall secure the meeting of the following requirements applicable to the instrumentation and control computer based systems and equipments important to nuclear safety:

- adequate regulations, procedures, and best practical experiences shall be used to secure the high quality of computer hardware and software; the level of their application shall be adequate to the importance of a given system in regard to nuclear safety
- the entire development process, including control, testing and commissioning of design change, shall be systematically documented and shall be reviewable
- experts not associated with the supplier shall evaluate the equipments in order to guarantee their high reliability
- a software-based common cause failure shall be considered
- protection shall be provided against accidental disruption of, or deliberate interference with, system operation.

3.7.1.1.2 The requirements for the instrumentation and control systems structure

The requirements for the instrumentation and control systems structure logically derive from the general requirement for defence in depth (this general requirement is specified in section "3.3.1.1.3 The concept of defence in depth" and it is further elaborated for instrumentation and control systems in section "3.7.1.1 The general requirements for the instrumentation and control systems"). The instrumentation and control systems design shall be designed as a functional whole in such a way that those systems shall meet the requirements for the functions and functionalities of instrumentation and control systems at a nuclear power plant, stipulated in valid legislation.

Each of the ETE3,4 units shall be equipped with autonomous instrumentation and control systems. The structure of these systems shall feature the subsystems specified below that shall meet the individual basic requirements for instrumentation and control systems:

- Reactor protection system - reactor trip system
- Reactor protection system - engineered safety feature systems
- Systems required for safe reactor shutdown
- Information systems important to safety
- Other information systems important to safety
- Control systems not required for safety
- Diverse instrumentation and control systems
- Data communication systems

The basic legislative requirements for all of these subsystems are specified in more detail within this Section of the ISAR.

3.7.1.2 REACTOR PROTECTION SYSTEM - REACTOR TRIP SYSTEM

3.7.1.2.1 System properties

Reactor protection systems will fulfill safety functions, and they simultaneously represent a part of the hierarchy of the defence in depth principle in nuclear safety of ETE3,4.

Protection reactor systems will consist of the reactor trip system (described in this Section) and of the engineered safety feature system (see Section 3.7.1.3).

The main task of the reactor protection system – reactor trip system – is to perform the functions in prevention of the reactor's operation outside of specified safety limits caused by equipment malfunction, operation mistakes, or other abnormal conditions.

3.7.1.2.2 Basic requirements for protection systems

References: Decree No. 195/1999 Coll., Article 17; SÚJB BN-JB-1.0 Safety Guide (88, 91); IAEA SSR 2/1, Req. 61, 6.32, WENRA App. E 10.9

The nuclear power plant will be equipped with protection systems that will meet the following requirements:

- Reactor protection systems will be capable to detect abnormal operation conditions and automatically initiate and turn off and disconnect or connect, as needed, relevant safety systems, including a powerful reactor shutdown safety system so that design criteria for abnormal operation will not be exceeded
- Reactor protection systems will be capable to detect accident conditions and, if needed, will be capable to actuate relevant safety systems designed to mitigate the impacts of such conditions
- In all the states considered in the nuclear facility design, the protection systems will be superior to the activities of control systems and nuclear facility operators so that interventions of control systems and operators could not devalue necessary effectiveness of safety systems; at the same time, the protection system shall provide instruments for back-up manual initialisation

of automatically initiated protection interventions from the main control room, as well as a back-up facility (a supplementary control room)

The design shall provide sufficient intervals between safety limits and specified limits for initiation of safety systems preventing undesirable frequent initiation of protection systems.

References: IAEA SSR 2/1 Req. 61, 6.33

The protection system design shall provide access to information on the status of the nuclear facility for the operators in such scope so that operators would be able to assess the effects of the automatic and manual interventions made by the protection system.

References: Decree No. 195/1999 Coll., Article 19(2); SÚJB BN-JB-1.0 Safety Guide (89)

The design will ensure that the protection system will be designed and set up in order to prevent exceeding of design limits, even under malfunction of the reactivity control system.

References: Decree No. 195/1999 Coll., Article 21(7); SÚJB BN-JB-1.0 Safety Guide (65); IAEA SSR 2/1, Req. 46, 6.12

Protection systems will enable periodical testing and online diagnostics that will ensure that the reactivity control and reactor shutdown systems have the required state and that necessary safety functions will be performed in all operation states, even under accident conditions.

References: Decree No. 195/1999 Coll., Article 18; SÚJB BN-JB-1.0 Safety Guide (86, 87, 88, 90)

The reactor safety systems will be designed with high functional reliability, redundancy, functional and physical diversity, and independence of individual channels so that:

- A single failure will not cause the loss of the protection system's function
- Disconnecting (putting out of operation) of a component or channel will not result in the reduction of the number of independent mutually redundant components or channels to one, unless the acceptable reliability of the protection system operation can be otherwise demonstrated in such case
- Possible threat to initiation or random initiation of safety systems' functions was minimized, even in case of common cause failure in protection systems unidentifiable in advance, such as program errors, sensor malfunctions, etc.

The protection systems will enable verification of operability of all of these systems' components during the reactor operation so that their failures that might reduce the functionality or redundancy of these systems will be reliably detected. Such verification will be based mainly on the continuously performed diagnostics of the component status, their periodical testing, and monitoring of redundant signals. For components with proven high reliability, it will be acceptable to verify their status only via periodical testing with the reactor shutdown, unless a practically feasible safe way of providing their testing during the unit operation exists.

The protection system will be designed to switch into the safe mode or another mode whose acceptability is justified and proved by an analysis if a failure of its components using the continuously automatically performed diagnostics is detected or the conditions disabling due performance of its safe functions occur. Such endangering conditions refer to e.g. switching the subsystems of the protection system off, loss of their power supply, or occurrence of predetermined unacceptable states of its operational ambient environmental conditions (extremely high or low temperatures, fire, extreme pressure, flooding with water or hitting with steam, high radiation, etc.).

References: IAEA SSR 2/1 Req. 21

Using sufficient means, the design will prevent mutual interactions between safety systems or between a redundant elements of a safety systems. Such means refer to, for example, physical separation, electrical separation, functional independence, and communication (data transfer) independence.

3.7.1.2.3 Requirements for the reactor trip system

The main task of the reactor trip system rests in automatic prevention of operation of the reactor under dangerous conditions by its shutdown whenever the reactor gets close to safety operation design limits.

References: Decree No. 195/1999 Coll., Article 17; SÚJB BN-JB-1.0 Safety Guide (88, 89)

The reactor shall be equipped with protection systems capable of shutting it down during both normal and abnormal operating conditions, as well as under accident conditions. The shutdown efficiency, speed, and margin (subcritical supply) of these systems shall be such that the specified design limits will not be exceeded even under malfunction of the reactivity control system.

References: Decree No. 195/1999 Coll., Article 21(8); SÚJB BN-JB-1.0 Safety Guide (66)

When the reactor is operational, a part of the reactor shutdown systems may be used for reactivity control or for neutron field shaping, provided that the margin for shutdown (bringing the reactor to the subcritical state) is maintained in the design continuously.

3.7.1.3 REACTOR PROTECTION SYSTEM - ENGINEERED SAFETY FEATURE SYSTEMS

3.7.1.3.1 System properties

Reactor protection systems will fulfill safety functions, and they simultaneously represent a part of the hierarchy of the defence in depth principle in nuclear safety of ETE3,4.

Protection reactor systems will consist of the reactor trip system (described in Section 3.7.1.2 above) and engineered safety feature system (described in this Section).

The main task of the reactor protection system – engineered safety feature systems is to perform the functions in the initiation of safety systems that automatically carry out protective interventions to mitigate consequences of design basis accidents.

3.7.1.3.2 Basic requirements for protection systems

The basic requirements for protection systems – engineered safety feature systems are consistent with requirements specified in Section "3.7.1.2.2 Basic Requirements for Protection Systems". With regard to the crucial importance of these basic requirements, the basic requirements for protection systems are specified again in this Section; however, within this Section, they are in the context of the engineered safety feature system.

Reference: Decree No. 195/1999 Coll., Article 17; SÚJB BN-JB-1.0 Safety Guide (88, 91); IAEA SSR 2/1, Req. 61, WENRA App. E 10.9

The nuclear power plant will be equipped with protection systems that will meet the following requirements:

- Reactor protection systems will be capable to detect abnormal operation conditions and automatically initiate and turn off and disconnect or connect, as needed, relevant safety systems, including a powerful reactor shutdown safety system so that design criteria for abnormal operation will not be exceeded
- Reactor protection systems will be capable to detect accident conditions and, if needed, will be capable to actuate relevant safety systems designed to mitigate the impacts of such conditions
- In all the states considered in the nuclear facility design, the protection systems will be superior to the activities of control systems and nuclear facility operators so that interventions of control systems and operators could not devalue necessary effectiveness of safety systems; at the same time, the protection system shall provide instruments for back-up manual initialisation of automatically initiated protective interventions from the main control room, as well as a back-up facility (a supplementary control room)

The design shall provide sufficient intervals between safety limits and specified limits for initiation of safety systems preventing undesirable frequent initiation of protection systems.

References: IAEA SSR 2/1 Req. 61, 6.33

The protection system design shall provide access to information on the state of the nuclear facility for the operators in such scope so that operators would be able to assess the effects of the automatic and manual interventions made by the protection system.

References: Decree No. 195/1999 Coll., Article 19(2); SÚJB BN-JB-1.0 Safety Guide (89)

The design will ensure that the protection system will be designed and set up in order to prevent exceeding of design limits, even under malfunctions of the reactivity control system.

Reference: Decree No. 195/1999 Coll., Article 21(7); SÚJB BN-JB-1.0 Safety Guide (65); IAEA SSR 2/1, Req. 46, 6.12

Protection systems will enable periodical testing and online diagnostics that will ensure that the reactivity control and reactor shutdown systems have the required state and that necessary safety functions will be performed in all operation states, even under accident conditions.

References: Decree No. 195/1999 Coll., Article 18; SÚJB BN-JB-1.0 Safety Guide (86, 87, 88, 90)

The reactor safety systems will be designed with high functional reliability, redundancy, functional and physical diversity, and independence of individual channels so that:

- A single failure will not cause the loss of the protection system's function
- Disconnecting (putting out of operation) of a component or channel will not result in the reduction of the number of independent mutually redundant components or channels to one, unless the acceptable reliability of the protection system operation can be otherwise demonstrated in such case
- Possible threat to initiation or random initiation of safety systems' functions was minimized, even in case of common cause failure in protection systems unidentifiable in advance, such as program errors, sensor malfunctions, etc.

The protection systems will enable verification of operability of all of these systems' components during the reactor operation so that their failures that might reduce the functionality or redundancy of these systems will be reliably detected. Such verification will be based mainly on the continuously performed diagnostics of the component status, their periodical testing, and monitoring of redundant signals. For components with proven high reliability, it will be acceptable to verify their status only via periodical testing with the reactor shutdown, unless a practically feasible safe way of providing their testing during the unit operation exists.

The protection system will be designed to switch into the safe mode or another mode whose acceptability is justified and proved by an analysis if a failure of its components using the continuously automatically performed diagnostics is detected or the conditions disabling due performance of its safe functions occur. Such endangering conditions refer to e.g. switching the subsystems of the protection system off, loss of their power supply, or occurrence of predetermined unacceptable states of its operational ambient environmental conditions (extremely high or low temperatures, fire, extreme pressure, flooding with water or hitting with steam, high radiation, etc.).

References: IAEA SSR 2/1 Req. 21

Using sufficient means, the design will prevent mutual interaction between safety systems or between a redundant elements of a safety systems. Such means refer to, for example, physical separation, electrical separation, functional independence, and communication (data transfer) independence.

3.7.1.3.3 Requirements for engineered safety feature systems

The main task of the engineered safety feature systems rests in execution of automatic protective interventions to mitigate consequences of design basis accidents.

References: Decree No. 195/1999 Coll., Article 17; SÚJB BN-JB-1.0 Safety Guide (46)

The design will be designed so that performance of safety system functions will be initiated and controlled automatically by protection systems or it will be implemented by passive means in such way that within 30 minutes of occurrence of the initiation event, the intervention of operating staff is not necessary. Any staff intervention required or necessary during 30 minutes after the occurrence of the postulated initiation event must be duly justified.

3.7.1.4 SYSTEMS REQUIRED FOR SAFE REACTOR SHUTDOWN

3.7.1.4.1 System properties

Instrumentation and control systems required for safe reactor shutdown will maintain the reactor in the safe state (hot zero power) for a longer period of time (usually several days) and, at the same time, these systems will fulfill functions of bringing the power plant in the cold shutdown state, if necessary.

The main task of systems required for safe reactor shutdown rests in provision of the measuring and controlling function that is necessary in order to maintain the safe state of the reactor shutdown and bring it to the cold shutdown state. No systems intended exclusively for keeping the unit in the cold shutdown are considered. Required functions for securing and maintaining the unit in the cold shutdown state and/or bringing it to the cold shutdown state will be provided by setting up selected power plant systems, which will normally ensure operation functions, including start up, shutdown, and protection functions.

3.7.1.4.2 Requirements for systems required for safe reactor shutdown

References: Decree No. 195/1999 Coll., Article 21(7, 8); SÚJB BN-JB-1.0 Safety Guide (65); IAEA SSR 2/1, Req. 46

Reactivity control and reactor shutdown systems will enable periodical testing and online diagnostics that will ensure that the reactivity control and reactor shutdown systems have the required state and that necessary safety functions will be performed in all operation states, even under accident conditions.

When the reactor is operational, a part of the reactor shutdown systems may be used for reactivity control or for neutron field shaping, provided that the margin for shutdown (putting the reactor into the subcritical state) is maintained in the design continuously.

3.7.1.4.3 Man-machine interface control requirements

References: Decree No. 195/1999 Coll., Article 16(1); SÚJB BN-JB-1.0 Safety Guide (83, 95); SSR 2/1, Req. 60

The nuclear power plant shall be equipped with control and information systems allowing to monitor, measure, record, further process, and properly control the operating parameters, technological processes, and systems important to nuclear safety and radiation and physical protection, and emergency preparedness during normal and abnormal operation and under accident conditions. These control and information systems shall provide the necessary visual and audio warnings informing of new or changed operating states, processes, and parameters that deviate from the admissible limits for the normal operation and may affect safety.

Control systems shall secure the maintaining of the operating parameters of technological systems important to nuclear safety in line with the design criteria, limits, and safe operating conditions.

3.7.1.4.4 Requirements for control rooms

Safe shutdown and switch to the cold shutdown state will be performed via control elements from the main control room, provided it is operable.

References: Decree No. 195/1999 Coll., Article 20 (1, 3); SÚJB BN-JB-1.0 Safety Guide (93, 95, 96, 97), IAEA SSR 2/1, Req. 65, 66

As the minimum, the nuclear power plant will be equipped with one main control room, from which it could be safely and reliably controlled and operated during normal and abnormal operation, as well as under accident conditions.

The control and information systems shall provide the necessary visual and audio warnings informing of new or changed operating states, processes, and parameters that deviate from the admissible limits for the normal operation and may affect safety. Operators will have available corresponding information for monitoring results of automatic interventions.

The design shall provide for application of ergonomic approaches in the control room's layout. Requirements for engineering psychology - ergonomics of the control room are specified in Section “3.18 Human factors engineering”.

The design shall be designed in a manner enabling shutdown, maintenance of the reactor in the safe state, and removal of residual heat, as well as monitoring of the power plant's state even if the main control room becomes inoperable. The design shall contain the respective back-up facility that may have the nature of a supplementary control room and shall be sufficiently physically and electrically separated from the main control room. In order to specify conditions for facilities and functionality of the instrumentation and control systems in the supplementary control room in case of accident conditions, it is admissible to use the probability approach.

Transfer and display of necessary data, which will be available to operators in the main control room, as well as to another suitable control and support worksite separated from both the main control room and the back-up facility (a supplementary control room) shall be provided, so that members of emergency support group, from which an emergency response can be directed,, are able to timely assess the state of the nuclear facility and critical safety functions under accident conditions.

Requirements for habitability of the main control room, as well as the back-up facility, are specified in Section “3.6.1.4 Habitability systems of control rooms”.

3.7.1.5 INFORMATION SYSTEMS IMPORTANT TO SAFETY

3.7.1.5.1 System properties

Information systems important to safety shall provide information necessary for safe operation under normal operation, transition, and especially accident and post-accident conditions.

The main task of the information systems important to safety rests in providing operating staff with information for manual interventions and monitoring and recording of operating parameters important to nuclear safety aspect during and after accident conditions.

3.7.1.5.2 Requirements for information systems important to safety

References: Decree No. 195/1999 Coll., Article 16 (1, 2, 3); SÚJB BN-JB-1.0 Safety Guide (83, 84, 95), IAEA SSR 2/1, Req. 32, 59, 5.56, 5.57, 6.31, WENRA App. F 3.1, 3.2

The nuclear power plant shall be equipped with control and information systems allowing to monitor, measure, record, further process, and control the operating parameters, technological processes, and systems important to nuclear safety and radiation and physical protection, and emergency preparedness during normal and abnormal operation and under accident conditions. The control and information systems shall provide the necessary visual and audio warnings informing of new or changed operating states, processes, and parameters that deviate from the admissible limits for the normal operation and may affect safety.

At the same time, the iformation systems shall also record, continuously in regular intervals or as needed, the values of the parameters that are important to nuclear safety of the nuclear facility, in accordance with safety analyses.

In case of occurrence of accident conditions, the information systems shall provide:

- Information regarding the current state of the nuclear plant, based on which the protective measures can be provided. This information shall contain at least information on parameters and system status, which can affect the course of the fission reaction and integrity of the fuel system of the primary circuit and the containment and related systems
- Basic information on the course of the accident and its recordings
- The information that enables the prognosis of the spread of radionuclides and ionising radiation to the vicinity of the nuclear power plant so that it will be possible to implement timely measures for the protection of the population.

References: Decree No. 195/1999 Coll., Article 20, SÚJB BN-JB-1.0 Safety Guide (97), IAEA SSR 2/1, Req. 65, 66

A selected set of outputs from the information systems important to safety specified within the design bases will be available at the main control room, a back-up facility, and a support and monitoring worksite in such a manner that these worksites would be able to fulfil their design functions.

3.7.1.6 OTHER SYSTEMS IMPORTANT TO SAFETY

3.7.1.6.1 System properties

Other systems important to safety include support systems whose reliable operation is absolutely necessary for full functionality of instrumentation and control systems important to nuclear safety.

The main task of these systems rests in provision of reliable power supply and ambient environmental conditions for instrumentation and control systems important to nuclear safety.

At the same time, this category includes safety interlocks implemented in control and protection systems whose main task rests in prevention of damage to the systems and equipments important to safety (unless such control systems and interlocks were already described in Sections 3.7.1.2 through 3.7.1.5).

3.7.1.6.2 The basic requirements for other systems that are important to safety

References: SÚJB BN-JB-1.0 Safety Guide (120)

The nuclear facility design shall define requirements for other auxiliary and supporting systems that provide important services or media to maintain operability of the systems and equipments important to nuclear safety, such as power supply, environmental conditions, and operating media (water, compressed air, fuel, lubrication, gases, etc.).

References: IAEA SSR 2/1 Req. 27, 5.42

Other systems important to safety are subject to all general requirements for systems important to nuclear safety applied in Section “3.7.1.1 for instrumentation and control systems”.

Supporting systems that will ensure operation of the systems and equipments, which form a part of the facilities important to nuclear safety, shall be classified adequately to safety importance of supporting facilities.

References: IAEA SSR 2/1 Req. 69

The design solution of supporting and other systems shall ensure that required performance of the systems will be in accordance with safety importance of components or systems, for which they provide supporting services.

References: IAEA SSR 2/1 Req. 27, 5.43

Malfunctions occurring in the support system will not be capable of simultaneously affecting redundant parts of safety systems or systems fulfilling diverse safety functions and compromising the capability of these systems to fulfill their safety functions.

3.7.1.6.3 Power supply of instrumentation and control systems important to safety

References: SÚJB BN-JB-1.0 Safety Guide (106)

The design must specify which systems and equipments important to safety require uninterrupted power supply and provide means of power supply for a sufficient time.

References: Decree No. 195/1999 Coll., Article 29 (2), SÚJB BN-JB-1.0 Safety Guide (103, 104), IAEA SSR 2/1 Req. 68, 6.43, 6.44, WENRA App. E 10.11

The design shall ensure that power supply for control, protection, and information systems important to safety will be solved as uninterrupted.

Electrical installation for power supply important control and protection systems of the primary circuit systems, systems for removal of residual heat, accident cooling systems, and containment systems shall enable power supply from an emergency source, i.e. it shall be backed-up without performance limitation for the time necessary for the systems' reliable functioning and regardless of whether own generators or the electrification system are in operation.

Emergency power supply systems shall be able to supply control, protection, and information systems important to nuclear safety with the corresponding power output in all operating states, as well as during design basis accidents, even in the case of their single failure and concurrent loss of external power supply.

3.7.1.6.4 Air conditioning and ventilation systems

References: SÚJB BN-JB-1.0 Safety Guide (119), IAEA SSR 2/1 Req. 73

The nuclear power plant shall be equipped with ventilation, air conditioning, and filtration systems that will ensure required conditions of areas where systems and equipments important to safety are located in all operating states, as well as during design basis accidents.

Design solution of ventilation, air conditioning, and filtration systems shall ensure that required environmental conditions of the environment for instrumentation and control systems important to nuclear safety will be fulfilled during all operating states, as well as during design basis accidents.

3.7.1.6.5 Safety interlocks

Safety interlocks include systems and interlocks important to safety that are not described in Sections 3.7.1.2 through 3.7.1.5.

References: IAEA SSR 2/1 Req. 40, 49, 5.70

Based on the executed analysis, safety interlocks that prevent over-pressurizing of the systems operating at the lower pressure when they are connected to the systems operating at the higher pressure or that prevent increase of pressure in the primary circuit in modes with low coolant temperature, shall be implemented in the instrumentation and control systems design.

Reference: Decree No. 195/1999 Coll., Article 19 (1); SÚJB BN-JB-1.0 Safety Guide (88), IAEA SSR 2/1, Req. 21, 64

The instrumentation and control systems design shall also implement interlocks that will prevent damage to systems and equipments important to safety (unless such control systems and interlocks are already described in Sections 3.7.1.2 through 3.7.1.5) and for separation of safety systems from other systems and for mutual separation of redundant and diverse systems during testing or maintenance.

3.7.1.7 CONTROL SYSTEMS NOT REQUIRED FOR NUCLEAR SAFETY

3.7.1.7.1 System properties

Control systems that are not required for nuclear safety will monitor and control the main and auxiliary technological systems of the electric power production process in the nuclear power plant that are not required from the nuclear safety aspect (they are not performing any safety function).

The main task of the control systems not required for nuclear safety rests in maintaining the specified operating parameters of technological systems and equipments, in line with the design criteria, limits, and safe operation conditions during the normal operation and, therefore, prevent abnormal operation or accident conditions. In case of abnormal operation or accident conditions, such systems can provide supporting functions that limit some undesirable consequences of abnormal operation or accident conditions.

3.7.1.7.2 Requirements for control systems not required for nuclear safety

The instrumentation and control systems form a part of the hierarchy structure of the nuclear power plant's instrumentation and control systems. They form an integral part of application of the general requirement for defence in depth in provision of nuclear safety of ETE3,4. Requirement for application of the general defence in depth requirement is specified in Section 3.7.1.1.

References: Decree No. 195/1999 Coll., Article 19 (1); SÚJB BN-JB-1.0 Safety Guide (88), IAEA SSR 2/1, Req. 22, 64

The protection and control systems will be separated so that a fault of the control systems does not affect the capability of the protection systems to perform the required safety function. The functionally necessary and practical connection of the protection and control systems will be limited as much as possible so that it does not have a significant effect on nuclear safety.

References: IAEA SSR 2/1 Req. 64, 6.38

If signals are in common by both a protection and any control systems, their separation shall be ensured, and those signals shall be classified as part of the protection system.

3.7.1.8 DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS

3.7.1.8.1 System properties

Diverse instrumentation and control systems shall provide diverse monitoring of parameters important to nuclear safety and implementation of diverse protection functions.

The main task of the diverse instrumentation and control systems rests in prevention of operation of the reactor under dangerous conditions by its shutdown in case of a common cause failure in the protection system and in automatic execution of applicable protective measures to mitigate consequences of the design basis accidents whenever the reactor gets close to safety operation design limits.

3.7.1.8.2 Requirements for diverse instrumentation and control systems

Reference: Decree No. 195/1999 Coll., Article 4 (1); SÚJB BN-JB-1.0 Safety Guide (44), IAEA SSR 2/1, Req. 24, 62, 6.34, 6.35, 6.36

The design of the nuclear power plant's instrumentation and control systems shall be sufficiently resistant against common cause failures of the instrumentation and control systems important to nuclear safety. These systems' design solution shall sufficiently apply principles of physical separation, functional isolation and independence, back-ups (redundancy), and physical and functional diversity.

Relevant diverse instrumentation and control systems shall be designed so that they fulfill requirements derived from the application of the general requirement for equipment quality of systems and equipments important to nuclear safety, in particular the principle of protection from multiple common cause failures. The requirement for application of the general requirement for quality is specified in Section 3.7.1.1.

References: Decree No. 195/1999 Coll., Article 18 (1), SÚJB BN-JB-1.0 Safety Guide (88)

Technical solutions that will ensure that protection systems will be designed with high functional reliability, multiplicity, and independence of individual channels shall be applied in the design.

The design of the instrumentation and control systems shall be solved in such a way as to minimize possible endangering of execution of safety functions in case of common cause failure in protection systems unidentifiable in advance, such as program errors, sensor malfunctions, etc.

References: SÚJB BN-JB-1.0 Safety Guide (92), IAEA SSR 2/1 Req. 63, 6.37, WENRA App. E 10.10

If computer based equipments are included in the protection systems design, an adequate certificate of quality, including its independent assessment, shall be required. If it is not possible to prove the reliable completion of designed safety functions of the protection systems sufficiently enough, the protection system's function shall be backed up on a different functional basis, in order to avoid common cause failures as much as possible.

3.7.1.9 DATA COMMUNICATION SYSTEMS

3.7.1.9.1 System properties

Data communication systems shall provide data communication within individual instrumentation and control systems, and also within complex hierarchical architecture of the nuclear unit's instrumentation and control system. Data communication devices usually form a part of individual instrumentation and control systems, between which communication buses are created.

The main task of the data communication systems rests in provision of transmission of technological and system data that is necessary for fulfillment of the main task of the instrumentation and control systems, i.e. for maintaining the specified operating parameters of technological systems, in line with the design criteria, limits, and safe operation conditions, and in case of systems important to safety, reliable fulfillment of all prescribed protection functions.

3.7.1.9.2 Requirements for data communication systems

The specified legislation does not contain the requirements for data communication systems.

This shall be specified in more detail within the next stages of the licensing documentation.

3.7.2 PROPERTIES OF THE INSTRUMENTATION AND CONTROL SYSTEMS DESIGN FOR THE PURPOSE OF THE PRELIMINARY ASSESSMENT

This section describes the properties of the design for the needs of a preliminary assessment of the design concept. The information for the specification of the design properties was based on the technical part of the tender documentation stipulating the requirements for safety and technical design of the future power plant design. This section includes partial assessment as to whether the tentative concept of the design segment in question complies with legislative requirements specified in Sections 3.7.1.1 through 3.7.1.9. The scope of this section includes identification and assessment of the general level of the requirements for the functions of the instrumentation and control systems, whereas particulars of the specific technological implementation shall be included in the design documentation of the selected supplier of the NPP, and the assessment will be performed within the next stage of the safety documentation.

The basic requirements for design of the instrumentation and control systems is derived from the general requirement for application of the defence in depth principle. (Requirements for the defence in depth concept are specified in Section "3.3.1.1.3 Concept of Defence in Depth" and requirements for the defence in depth implementation process are specified in Section "3.3.1.1.4 Approach to Implementation of Defence in Depth". For the instrumentation and control systems, the defence in depth principle is specified in Section "3.7.1.1 The General Requirements for the Instrumentation and Control Systems").

The instrumentation and control systems shall be designed so as to ensure meeting the basic safety requirements and principles described in Section 3.3.1. These include, in particular, the principle of defence in depth and the design criteria specified for the systems, structures, and components within the design requirements, including the relevant design requirements based on the conditions of the site, as specified in Chapter 2 and summarized for the most critical ones in Section 2.10. Legislative requirements derived from specified legislative supporting documents for processing of ISAR are summarized in Sections 3.7.1.1 through 3.7.1.9.

The general principles of the instrumentation and control system design valid for all the subsystems of the instrumentation and control system, that are considered in the design solution, are specified in Section 3.7.2.1.

The design shall include the subsystems of the instrumentation and control system, specified in Sections 3.7.2.2 through 3.7.2.9, whereas the particular method of their implementation will be specified in the draft of the particular design solution of the nuclear power plant. The design solution selected for implementation does not have to include all systems, structures, and components specified in this Section; however, if the selected design utilizes them, they shall fulfill applicable requirements specified in this Section.

3.7.2.1 INSTRUMENTATION AND CONTROL SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

The nuclear power plant design shall include instrumentation and control systems that provide monitoring and control functions of technology systems supporting power production at the nuclear power plant, especially systems important to nuclear safety.

These systems shall be designed in accordance with requirements of mechanical-technical systems, electrical and construction systems, and in accordance with specified requirements for fulfillment of specified safety functions in such a way that the specified level of fulfillment of the defence in depth principle, the resistance against common cause failure, and fulfillment of a single failure criterion will not decrease.

Purpose and importance

The required solution of the instrumentation and control systems shall ensure that such systems will maintain the specified operating parameters of technological systems, in line with the design criteria, limits, and safe operation conditions, including the reliable operation of all specified safety functions in the case of systems important to safety.

The design solution of the instrumentation and control systems shall make sure the systems contribute to the maintaining of the necessary properties concerning power generation, manoeuvrability, etc. while maintaining the required safety levels during normal and abnormal operations and under emergency conditions.

The basic principles of the instrumentation and control system design solution

The required design solution of the instrumentation and control system shall secure the specification of such instrumentation and control system means that support all the functions that must be featured by the instrumentation and control system in accordance with the specified general requirements and safety goals.

The instrumentation and control system design shall be created during a complex process securing the respecting of all the requirements that must be met by the instrumentation and control system, their implementation into the design and their maintenance during the power plant's whole lifetime.

The instrumentation and control system design shall be processed completely in accordance with the design processes of the power plant's other parts / systems.

The required design solution shall lead to standardized selection of technical means of the instrumentation and control system that shall reasonably limit the group of applied systems and components in order to facilitate maintenance and training, to simplify spare parts requirements and to support easy replacement and upgrading of the I&C equipment the instrumentation and control systems.

The required design solution shall secure such a instrumentation and control system design and implementation that shall enable simple and easy maintenance and system and component replacements.

The design shall focus on the modularity of applied systems (HW and SW). The instrumentation and control systems shall feature open architecture and their interfaces shall be standardized (i.e. based on internationally accepted standards).

The supplier of the instrumentation and control systems shall provide instrumentation and control systems obsolescence program (down to the individual component level) in which it shall define the strategy to support the overall design lifetime of the instrumentation and control systems.

Design solution principles - General principles:

Defence in depth

The requested design solution of the instrumentation and control systems shall make sure that their structure follows the defence in depth principles and that the required independence is reached between the individual established lines of defence. The instrumentation and control systems shall be segmented according to their importance to nuclear safety while meeting the general requirements for equipment standardization.

During the instrumentation and control system design, the defence in depth principle shall be technically implemented via a hierarchical structure of instrumentation and control systems (see Section "3.7.1.1.2 The requirements for the instrumentation and control systems structure") that shall feature the necessary levels of independence, separation, functional isolation, redundancy, and physical and functional diversity.

Quality requirements

The required design solution shall secure the implementation of the following basic requirements for the instrumentation and control systems' quality:

Redundancy

The redundancy principle that will secure fulfillment of the single failure criterion for the systems performing safety functions shall be applied in the instrumentation and control systems. The supplier shall apply this principle even for the systems that affect reliability of the power plant operation.

The requested design solution shall technically ensure fulfillment of the minimum redundancy requirement, in accordance with the N+2 criterion (i.e. fulfillment of the single failure criterion plus satisfactory service/maintenance) in the instrumentation and control systems important to the nuclear safety.

When proposing a particular redundancy degree, the supplier of the instrumentation and control systems shall take into account requirements regarding the single failure criterion and results of the preliminary safety assessment.

When making this proposal, the supplier shall also take into account the fact that requirements for redundancy of passive systems may be less than those for active systems.

Diversity

The diversity principle whose implementation will prevent occurrence of common cause failures shall be applied in design of the instrumentation and control systems. The supplier of the instrumentation and control systems shall propose utilization of such manner and degree of diversity that will correspond with the importance of safety functions to be performed by the relevant system or component.

Properties of design of the instrumentation and control systems providing functions of diverse monitoring of parameters important to nuclear safety and implementation of diverse protection functions are described in Section "3.7.2.8 Diverse instrumentation and control systems in the Tentative Design Concept".

Proven technology

In the instrumentation and control system design, modern and advanced technology (digital microprocessor technology, programmable controllers, distributed computer systems, fiber optic, modern displays, etc.) shall be preferred for individual systems. The supplier shall evaluate the instrumentation and control system technologies that are considered from the aspect of assessment of the experience of such technology at other power plants or relevant industrial applications and shall use for the ETE3,4 design only those that are fully proven to fulfill specific functional, performance, and qualification conditions.

The instrumentation and control system design solution shall not utilize prototype solutions, devices at the end of their life cycle and technologies, that have not been proven in corresponding applications.

Achievement of design objectives

The requested instrumentation and control system design solution shall ensure that all specified design objectives will be achieved to a high quality and within the given time and cost frames.

All the design objectives shall be achieved during the whole instrumentation and control system life cycle.

Classification

In the design solution proposal for the whole power plant, there shall be propose, describe and apply an appropriate and consistent classification system, including associated safety requirements (seismic classification, environmental qualification, associated construction requirements, and quality assurance requirements). The supplier shall submit justification of the proposed classification system and shall be fully responsible for to comply with licensing requirements of the State Office for Nuclear Safety (SÚJB).

It is recommended to classify the instrumentation and control systems and equipments in accordance with the ČSN IEC 61226 standard.

The design solution proposal shall contain classification of the instrumentation and control systems and equipments into safety classes, in accordance with Decree No. 132/2008 Coll. [L. 258].

The instrumentation and control systems design for every defined systems and equipments category shall stipulate specific requirements as for robustness, qualification, and reliability of the instrumentation and control systems, including requirements for parameters of such systems.

General requirements for classification of individual systems, structures, and components of a power plant are specified in Section "3.3.2 Classification of structures, components and systems".

Equipment failure protection

The design solution shall contain division of the instrumentation and control systems and equipments into categories from the aspect of fulfillment of safety functions, and requirements for seismic qualification and environmental qualification shall be specified for individual categories.

General requirements for qualification of individual systems, equipments and components of a power plant are specified in Section "3.3.1.1.13 Equipment Qualification".

The instrumentation and control system design will be designed in such a way that location of the instrumentation and control systems and equipments corresponds with design properties of the surrounding environment, including EMC in that particular location.

Everywhere where it is reasonably practicable, the Fail-safe principle shall be incorporated into the instrumentation and control system design solution. Therefore, the design will ensure that in case of a system or component malfunction, the power plant will pass into the safe state without a requirement to initiate any actions.

Separation of control and protection systems

The required design solution shall ensure application of requirements for independence and functional separation of instrumentation and control systems fulfilling protection functions from other instrumentation and control systems. The design solution shall prevent mutual interference between protection and control systems by means of not connecting them or using such technical means that will ensure appropriate functional isolation.

If some input signals are used common by both protection and control systems simultaneously, the design solution shall ensure and prove that all safety requirements for protection systems have been met.

Use of computer systems in systems important to safety

The requested design solution of instrumentation and control systems executing functions important to nuclear safety, where software based systems are used, shall involve diverse instrumentation and control systems so that the requirements of reliability targets and defence in depth are met.

The requested design solution shall ensure sufficient protection of digital computer and communication systems and networks against cyber attack. In order to fulfill this requirement, the proposed design solution shall utilize the most rigorous procedures, methodologies, and rules that are available in this area.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for systems, structures and components, and safety and technological functions of the instrumentation and control systems specified in Section "3.7.2.1 Instrumentation and control systems in the tentative design concept" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll.: Article 3 (1), Article 4 (1)(2), Article 7, Article 16 (1), Article 19 (1) of document IAEA SSR 2/1 Req. 7 (4.9, 4.10, 4.11), 9 (4.14), 22, 23, 24, 25, 26, 29, 30 (5.48), 39, 59, 60, 63 (6.37), 64 (6.38) and the SÚJB BN-JB-1.0 Safety Guide (5, 6, 7, 10, 11, 12, 19, 21, 41, 42, 43, 83, 85, 88, 92, 95).

3.7.2.2 REACTOR PROTECTION SYSTEM - REACTOR TRIP SYSTEM IN THE TENTATIVE DESIGN CONCEPT

The nuclear power plant instrumentation and control system shall include reactor protection systems as its the most important part.

Such protection systems shall fulfill safety functions, and at the same time, they represent a part of the defence in depth principle's hierarchy in provision of nuclear safety of ETE3,4. Reactor protection systems shall consist of the reactor trip system (the design tentative concept is specified in this Section) and the engineered safety feature system (for the design tentative concept see Section 3.7.2.3).

Purpose and importance

The requested design solution shall ensure that the reactor protection system – reactor trip system will carry out protection functions in prevention of the reactor's operation outside of specified safety limits caused by equipment malfunction, operation mistakes, or other abnormal conditions.

Design solution principles

The design solution of the reactor protection system – the reactor trip system shall ensure fulfillment of the general requirements for instrumentation and control systems defined in Section "3.7.1.1 The general requirements for the instrumentation and control systems". The partial assessment of the tentative concept of the instrumentation and control system design from the aspect of fulfillment of general requirements is shown in Section "3.7.2.1 Instrumentation and control systems in the design's tentative concept".

In addition to aforementioned general requirements, the requested design solution of protection systems shall be solved so that such systems fulfill other requirements for functions of the protection systems specified below.

The requested nuclear power plant design solution shall include protection systems consisting of electrical and mechanical systems and aquipment and circuits (from input sensors to action components). Such systems shall also include protection signals creating circuits.

Design solution of the protection systems shall ensure that protection systems will detect accidents and accident conditions and will fulfill the following safety functions:

- automatic reactor trip
- automatic actuation of engineered safety feature systems

Design solution of the protection systems shall ensure that in addition to safety functions, the protection systems shall provide the following functions:

- Initiation of the turbine-generator trip
- Generation of signals used for interlock control or protective actions depending on the power plant's operating conditions

The protection systems shall be designed in the design as autonomous – if selected parameters exceed setpoints, the protection systems shall automatically without an operator's intervention execute activities required by the design, i.e. reactor trip and/or initiation of engineered safety feature system.

The design solution of protection systems shall be derived from functional requirements of mechanical-technical, electrical, and construction systems, and it shall be in accordance with specified requirements for execution of safety functions. These functional requirements are specified in Section "3.6 Safety Systems".

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for systems, structures, and components, and safety and technological functions of the instrumentation and control systems specified in Section "3.7.2.2 Reactor protection system – reactor trip system in the tentative design concept" creates preconditions for meeting the legislative requirements specified in the partial preliminary assessment of Section 3.7.2.1 and other specific requirements of Decree No. 195/1999 Coll. [L. 266], Article 17, Article 18, Article 19 (2), Article 21 (7)(8) of document IAEA SSR 2/1 [L. 252] Req. 21, 46 (6.12), 61 (6.32, 6.33) and SÚJB BN-JB-1.0 Safety Guide [L. 276] (65, 66, 86, 87, 88, 89, 90, 91).

3.7.2.3 REACTOR PROTECTION SYSTEM - ENGINEERED SAFETY FEATURE SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

The nuclear power plant instrumentation and control system shall include reactor protection systems as its the most important part.

Such protection systems shall fulfill safety functions and, at the same time, they represent a part of the defence in depth principle's hierarchy in provision of nuclear safety of ETE3,4. Reactor protection systems shall consist of the reactor trip system (the design tentative concept is specified in Section 3.7.2.2) and the engineered safety feature system (the design tentative concept is specified in this Section).

Purpose and importance

The requested design solution shall ensure that the reactor protection system – engineered safety feature systems will initiate safety systems, which automatically execute protective interventions to mitigate consequences of design basis accidents.

Design solution principles

The design solution of the reactor protection system – engineered safety feature systems shall ensure fulfillment of the general requirements for instrumentation and control systems defined in Section "3.7.1.1 The general requirements for the instrumentation and control systems". The partial assessment of the tentative concept of the instrumentation and control system design from the aspect of fulfillment of general requirements is shown in Section "3.7.2.1 Instrumentation and control systems in the design's tentative concept".

The basic requirements for protection systems – engineered safety feature systems are consistent with requirements specified for the protection system – the reactor trip system. The design solution principles are specified in Section 3.7.1.2 and the partial assessment of the tentative concept of the reactor trip system design from the aspect of fulfillment of these basic requirements is specified in Section "3.7.2.2 The reactor protection system – the reactor trip system in the tentative design concept".

In addition to aforementioned general requirements and basic requirements for protection systems, the requested design solution of the engineered safety feature system shall be solved so that such systems fulfill other requirements for functions specified below.

The requested nuclear power plant design solution shall ensure that in case of accident activation of the protection system, the required actions will be executed automatically for no less than 30 minutes without the need for an operator intervention from the main control room. Any intervention of operating staff, that must be performed within this time period, shall be identified and justified in the design.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for systems, structures, and components, and safety and technological functions of the instrumentation and control systems specified in Section "3.7.2.3 Reactor protection system - engineered safety feature systems in the tentative design concept" creates preconditions for compliance with all the legislative requirements specified in the partial preliminary assessment of Section 3.7.2.1 and other specific requirements of Decree No. 195/1999 Coll. [L. 266], Article 17, Article 18, Article 19 (2), Article 21 (7) of document IAEA SSR 2/1 [L. 252] Req. 21, 46 (6.12), 61 (6.33) and SÚJB BN-JB-1.0 Safety Guide [L. 276] (46, 65, 86, 87, 88, 89, 90, 91).

3.7.2.4 SYSTEMS REQUIRED FOR SAFE REACTOR SHUTDOWN IN THE TENTATIVE DESIGN CONCEPT

The nuclear power plant instrumentation and control system shall include systems performing functions required for safe reactor shutdown.

Applied design requirements do not consider any systems intended exclusively for keeping and/or bringing the unit in the cold shutdown state. Design-required functions for securing and maintaining the unit in the cold shutdown state and/or bringing it to the cold shutdown state will be provided by selected power plant systems, which will normally ensure operation functions, including start up, shutdown, and protection functions.

Purpose and importance

The required design solution shall ensure that information and control systems required for safe reactor shutdown will maintain the reactor in the safe mode (hot zero power) for a longer period of time (usually several days) and, at the same time, these systems will fulfill functions of bringing the power plant in the cold shutdown conditions, if necessary.

The design solution of the systems that will fulfill functions of securing and maintaining the unit in the cold shutdown state and/or bringing it to the cold shutdown state shall ensure fulfillment of the general requirements for instrumentation and control systems defined in Section "3.7.1.1 The general requirements for the instrumentation and control systems". The partial assessment of the tentative concept of the instrumentation and control system design from the aspect of fulfillment of general requirements is shown in Section "3.7.2.1 Instrumentation and control systems in the design's tentative concept".

In addition to the aforementioned general requirements, the requested design solution of systems fulfilling functions required for safe reactor shutdown shall be solved so that such systems fulfill other requirements for functions specified below.

Requirements for control rooms

The nuclear power plant design shall include a main control room and a supplementary control room for every unit.

The requested design solution of the main control room shall ensure that instruments enabling safe operation of a nuclear unit under normal conditions will be available in the main control room and, at the same time, these instruments will enable to switch a unit to a safe state after occurrence of accident conditions or design extension conditions.

Note:
The design shall also include a supplementary control room that will be utilized when a main control room is not accessible. In order to specify conditions for equipment and functionality of the instrumentation and control systems in the supplementary control room in case of accident conditions, it is admissible to use the probability approach for assessment of simultaneous occurrence of the main control room being inhabitable and an accident.

The design solution of workplaces, their layout, a method of access to information and control instruments, and presentation of information shall be designed in accordance with principles of engineering psychology and ergonomics. The requested design solution of control rooms in respect of engineering psychology and ergonomics is specified in Section "3.18.2 Properties of the human factors engineering design for the purposes of preliminary assessment".

The requested design solution shall ensure that the process of design of the Man-Machine Interface (MMI) will consider all required functions and tasks and will include them to the detailed MMI design.

The Man-Machine Interface design shall be a result of a complex process, in which all required functions and tasks, operation aspects, and safety and comfort of workers will be taken into account.

The requested design solution of habitability and environment of both the operating and supplementary control rooms is specified in section "3.6.2.4 Habitability systems of the control rooms in the tentative design concept".

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for systems, structures, and components, and safety and technological functions of the instrumentation and control systems specified in Section "3.7.2.4 Systems required for safe reactor shutdown in the tentative design concept" respects all legislative requirements specified in a partial assessment of Section 3.7.2.1 and other specific requirements of Decree No. 195/1999 Coll. [L. 266], Article 16 (1), Article 21 (7)(8), Article 20 (1), (3) of document IAEA SSR 2/1 [L. 252] Req. 46, 60, 65, 66 and SÚJB BN-JB-1.0 Safety Guide [L. 276] (65, 66, 83, 93, 95, 96, 97).

3.7.2.5 INFORMATION SYSTEMS IMPORTANT TO SAFETY IN THE TENTATIVE DESIGN CONCEPT

The nuclear power plant instrumentation and control system shall include information systems performing functions important to nuclear safety.

Purpose and importance

The requested design solution shall ensure that information systems important to safety shall provide information necessary for safe operation during normal, transition, and especially accident and post-accident conditions. Their main task shall rest in providing operating staff with information for manual interventions and monitoring and recording of operating parameters important to nuclear safety during and after accident conditions.

Design solution principles

The design solution of the information systems important to safety shall ensure fulfillment of the general requirements for instrumentation and control systems defined in Section "3.7.1.1 The general requirements for the instrumentation and control systems". The partial assessment of the tentative concept of the instrumentation and control system design from the aspect of fulfillment of general requirements is shown in Section "3.7.2.1 Instrumentation and control systems in the design's tentative concept".

In addition to the aforementioned general requirements, the requested design solution of information systems important to nuclear safety shall be solved so that such systems fulfill other requirements for functions specified below.

The design solution proposal shall contain analysis of information flow and analysis of processing requirements. Such analysis shall be executed for determination of information requests, information flows, and requests for data processing in instrumentation and control systems to make it possible technically accomplish all required functions in the design, including decision making and operation activities. The requested design solution shall ensure that instrumentation and control systems will meet all the requirements specified based on this analysis.

The requested design solution shall include instrumentation and instrumentation devices for monitoring of accident and post-accident states. Such instrumentation shall be implemented by means providing monitoring of fulfillment of all safety functions and, at the same time, timely detection of the risk of loss of safety functions. On this basis, measures for minimization and prediction of released radioactive substances to the primary containment and to the surroundings.

The design shall ensure and prove that instrumentation of the accident and post-accident monitoring will fulfill its function under the design-specified conditions.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for systems, structures, and components, and safety and technological functions of the instrumentation and control systems specified in Section "3.7.2.5 Information systems important to safety in the tentative design concept" creates preconditions for compliance with all the legislative requirements specified in the partial preliminary assessment of Section 3.7.2.1 and other specific requirements of Decree No. 195/1999 Coll. [L. 266], Article 16 (1)(2)(3), Article 20 of document IAEA SSR 2/1 [L. 252] Req. 32, 59 (5.56, 5.57, 6.31), 65, 66 and SÚJB BN-JB-1.0 Safety Guide [L. 276] (83, 84, 95, 97).

3.7.2.6 OTHER SYSTEMS IMPORTANT TO SAFETY IN THE TENTATIVE DESIGN CONCEPT

In addition to systems defined in Sections 3.7.2.2 through 3.7.2.5, the nuclear power plant instrumentation and control system shall also include other systems performing functions important to nuclear safety. Such systems include supporting systems that provide reliable power supply and environmental conditions for instrumentation and control systems important to nuclear safety. At the same time, this category includes safety interlocks implemented in control and protection systems whose main task shall rest in prevention of damage to systems and equipments important to safety (unless such control systems and interlocks were already described in Sections 3.7.2.2 through 3.7.2.5).

Purpose and importance

The requested design solution shall ensure that supporting systems shall provide reliable power supply and environmental conditions for instrumentation and control systems important to nuclear safety. The design shall further ensure that safety interlocks included in this category of systems will efficiently prevent damage to the systems and equipments important to safety.

Design solution principles

The design solution of the other systems important to safety shall ensure fulfillment of the general requirements for instrumentation and control systems defined in Section "3.7.1.1 The general requirements for the instrumentation and control systems". The partial assessment of the tentative concept of the instrumentation and control system design from the aspect of fulfillment of general requirements is shown in Section "3.7.2.1 Instrumentation and control systems in the design's tentative concept".

In addition to the aforementioned general requirements, the requested design solution of other (supporting) systems shall be solved so that such systems fulfill other requirements for functions specified below.

The design solution of power supply for the instrumentation and control systems shall ensure that quality of power supply for such systems will fully correspond with requirements for safety functions performed by powered instrumentation and control systems and equipments.

Properties of the design of power supply for instrumentation and control systems is specified in Section “3.8.2 Properties of the electrical system design for the purpose of preliminary assessment”.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for systems, structures, and components, and safety and technological functions of the instrumentation and control systems specified in Section "3.7.2.6 Other systems important to safety in the tentative design concept" creates preconditions for compliance with all the legislative requirements specified in the partial preliminary assessment of Section 3.7.2.1 and other specific requirements of Decree No. 195/1999 Coll. [L. 266], Article 19 (1), Article 29 (2) of document IAEA SSR 2/1 [L. 252] Req. 21, 27 (5.42, 5.43), 40, 49 (5.70), 64, 68 (6.43, 6.44), 69, 73 and SÚJB BN-JB-1.0 Safety Guide [L. 276] (88, 103, 104, 119, 120).

3.7.2.7 CONTROL SYSTEMS NOT REQUIRED FOR NUCLEAR SAFETY IN TENTATIVE DESIGN CONCEPT

The nuclear power plant instrumentation and control system design shall also include instrumentation and control systems that are not required from the nuclear safety aspect (they are not performing any safety function). Such systems shall ensure monitoring and control of such technological systems of the electric power production process in the nuclear power plant that are not required from the nuclear safety aspect.

Such instrumentation and control systems shall form a part of the hierarchy structure of the power plant's instrumentation and control systems and shall form an integral part of the application of the general requirement for defence in depth in the area of provision of nuclear safety in ETE3,4.

Purpose and importance

The requested design solution shall ensure that such control systems will perform functions in respect of maintaining the specified operating parameters of technological systems, in line with the design criteria, limits, and safe operation conditions during the normal operation and, therefore, prevent abnormal operation or accident conditions. In case of abnormal operation or accident conditions, such systems can provide supporting functions that limit some undesirable consequences of abnormal operation or accident conditions.

Design solution principles

The design solution of the other systems important to safety shall ensure fulfillment of the general requirements for instrumentation and control systems defined in Section "3.7.1.1 The general requirements for the instrumentation and control systems". The partial assessment of the tentative concept of the instrumentation and control system design from the aspect of fulfillment of general requirements is shown in Section "3.7.2.1 Instrumentation and control systems in the design's tentative concept".

In addition to the aforementioned general requirements, the requested instrumentation and control system design solution shall be solved so that such systems fulfill other requirements for functions specified below.

The requested design solution shall ensure application of requirements as for independence and functional separation of instrumentation and control systems fulfilling protection functions from other instrumentation and control systems. The design solution shall prevent interconnection of protection and control systems using such technical means that will ensure appropriate functional isolation.

If some input signals are used common by both protection and control systems simultaneously, the design solution shall ensure and prove that all safety requirements for protection systems have been met.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for systems, structures, and components, and safety and technological functions of the instrumentation and control systems specified in Section "3.7.2.7 Control systems not required for nuclear safety in the tentative design concept" creates preconditions for compliance with all the legislative requirements specified in the partial preliminary assessment of Section 3.7.2.1 and other specific requirements of Decree No. 195/1999 Coll. [L. 266], Article 19 (1) of document IAEA SSR 2/1 [L. 252] Req. 22, 64 (6.38) and SÚJB BN-JB-1.0 Safety Guide [L. 276] (88).

3.7.2.8 DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

The nuclear power plant instrumentation and control system design shall include systems that shall provide diverse monitoring of parameters important to nuclear safety and implementation of diverse protection functions.

Such diverse systems shall form a part of the hierarchy structure of the power plant's instrumentation and control systems and shall form an integral part of the application of the general requirement for defence in depth in the area of provision of nuclear safety in ETE3,4.

Purpose and importance

The requested design solution shall ensure that in case of a common cause failure in the protection system, the diverse systems will prevent operation of the reactor in a dangerous state by its shutdown and automatic execution of applicable protective measures to mitigate consequences of the design basis accidents whenever the reactor gets close to safety operation design limits. At the same time, the diverse systems shall provide diverse monitoring of parameters important to nuclear safety.

Design solution principles

The design solution of the diverse instrumentation and control systems shall ensure fulfillment of the general requirements for instrumentation and control systems defined in Section "3.7.1.1 The general requirements for the instrumentation and control systems". The partial assessment of the tentative concept of the instrumentation and control system design from the aspect of fulfillment of general requirements is shown in Section "3.7.2.1 Instrumentation and control systems in the design's tentative concept".

In addition to the aforementioned general requirements, the requested design solution of diverse instrumentation and control systems shall be solved so that such systems fulfill other requirements for functions specified below.

The requested design solution of instrumentation and control systems executing functions that are important to nuclear safety shall include diverse instrumentation and control systems so that the requirements of reliability targets and defence in depth are met. The supplier may fulfill these requirements by combination of active and passive systems and components.

The supplier shall propose a technical solution of the instrumentation and control systems taking into account the diversity principle especially where software based instrumentation and control systems, including MMI, will be utilized. This design solution shall contain a "hard-wired" back-up system or diverse software and shall be proposed in cases when it will be necessary for achieving required reliability of the instrumentation and control systems.

The design solution must contain the particular decision on utilization of diversity, selection of a diversity type, or the decision not to use diversity.

The design solution shall include proof that diversity is in fact achieved in the implemented design and will be preserved for the whole duration of the system's life cycle.

Objectives of the requested proof shall be achieved in the design solution in a manner where the supplier shall review the instrumentation and control system design as a complex whole while applying the diversity principle, with the objective to avoid areas of potential homogeneity, such as material, components, similar production processes, similar software, or similarities of work principles or common supporting features.

If the protection systems based on digital computer systems ("computer based") are used in the design and it is not possible to prove reliability of execution of designed safety functions of such systems with sufficient credibility, the design solution of the protection systems shall contain a back-up diverse system (including the diverse MMI, sensors, and actuators) based on "non-computer based" technology.

The design solution of implementation of functional, software, and instrumental diversity shall be based on the deterministic approach, together with assessment of risk of common cause failures and in accordance with PSA.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for systems, structures, and components, and safety and technological functions of the instrumentation and control systems specified in Section "3.7.2.8 Diverse instrumentation and control systems in the tentative design concept" creates preconditions for compliance with all the legislative requirements specified in the partial preliminary assessment of Section 3.7.2.1 and other specific requirements of Decree No. 195/1999 Coll. [L. 266], Article 4 (1), Article 18 (1) of document IAEA SSR 2/1 [L. 252] Req. 24, 62 (6.34, 6.35, 6.36), 63 (6.37) and SÚJB BN-JB-1.0 Safety Guide [L. 276] (42, 88, 92).

3.7.2.9 DATA COMMUNICATION SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

The preliminary assessment of the design concept in the area of data communication systems has not been carried out because the specified range of the binding legislation does not delineate any requirements in the assessed area.

3.7.3 COMPREHENSIVE PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT

The design of ETE3,4 shall meet the safety requirements specified in Section "3.3.1.1 Basic legislative requirements for provision of safety", as well as the requirements specified in Sections 3.7.1.1 through 3.7.1.9, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and its implementing decrees regarding nuclear safety, radiation protection, and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270] documents.

To meet the aforementioned requirements, the ETE3,4 design shall use systems, structures, and components, and their equipment integrated within the instrumentation and control systems that will ensure (with adequate reliability and resistance) qualitatively and quantitatively adequate monitoring, control, and protection functions of technical systems (especially mechanical-technical, construction, and electrical systems), including systems important to nuclear safety, in accordance with specified requirements for fulfillment of required safety functions.

The principles of the design solution described in Section "3.7.2 Properties of the instrumentation and control systems design for the purpose of preliminary assessment" were set up based on the requirements submitted by the applicant for a licence applied to the potential suppliers of the nuclear facility within the tender, and they make up the concept of the design solution for this part of the design. The partial assessments performed proved that the expected design of the instrumentation and control systems creates preconditions for compliance with the relevant requirements for systems, structures, and components and safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], SÚJB BN-JB-1.0 Safety Guide [L.

276], document IAEA SSR 2/1 [L. 252], and the document WENRA [L. 27]. The instrumentation and control systems suit the implemented preliminary assessment of the design concept.

The particular method of technical implementation of individual requirements for operation of the instrumentation and control systems specified in Section 3.7.2 shall be specified in detail only in the design documentation of the selected nuclear power plant supplier.

3.8 ELECTRICAL SYSTEMS

This comprehensive part of the Initial Safety Analysis Report is structured into three basic sections:

The opening Section 3.8.1 summarizes, analyses, and specifies the basic legislative requirements for electrical systems, including specific requirements for provided power supply for own consumption, back-up of power supply systems, and emergency power supply sources. This section also includes requirements for the external electrical system, electrical systems inside the power plant, and solution of events such as "Station Blackout" (SBO).

Section 3.8.2, which follows, contains a description and specification of the basic requirements for the functions of the electrical systems specified in the opening Section 3.8.1 in the form of "envelope" applied design requirements for technical subsystems of the electrical systems within all the relevant designs involved in the current tender procedure. The section's goal is to formulate the design's general properties for the purposes of partial preliminary assessment.

The final Section 3.8.3 contains a comprehensive preliminary assessment of the concept of the electrical systems, summarizing the conclusions of the partial preliminary assessments presented in Section 3.8.2. The assessment of the summarized requirements contains a tentative design concept assessment required by the law.

The applicant shall use the next section of the licence documentation in order to provide assessment and supporting information on the selected design that will allow the assessment of the electrical systems' ability to fulfill the specified safety functions during the nuclear unit's whole lifetime under any operating conditions, including accident conditions. The section shall also be supplemented with detailed information at the depth and in the structure described in RG 1.206 [L. 275].

3.8.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE ELECTRICAL SYSTEMS

The electrical systems lead the output out to the electrification system and also provide normal, stand-by, and emergency auxiliary power supply for house load, including systems important to nuclear safety. The main task of the electrical systems rests in provision of the required qualitative and quantitative power supply parameters that will enable reliable operation of supplied technological systems, and in the case of systems important to nuclear safety, it will also enable reliable fulfillment of all required safety functions.

3.8.1.1 GENERAL REQUIREMENTS FOR THE ELECTRICAL SYSTEMS

This Section contains general requirements for electrical systems as a whole and general requirements for provision of auxiliary power supply , back-up of power supply systems, and emergency power supply sources.

References: SÚJB BN-JB-1.0 Safety Guide (99)

Nuclear installations containing a nuclear reactor shall have available an external and internal auxiliary power supply system so that installation important to nuclear safety is able to perform specified functions, whereas every one of these systems (provided the other one is out of operation) on its own accord shall have sufficient reliability and capacity so that:

- Design limits of the active zone and the pressure circuit of the reactor in normal, as well as abnormal operation, and safety functions in the conditions of the basic design basis accidents are ensured,
- Their failures (if any) have as little impact on the reactor's safety as possible.

References: SÚJB BN-JB-1.0 Safety Guide (106)

The design shall specify which systems important to safety require uninterrupted power supply and provide means of power supply for a sufficient time.

References: SÚJB BN-JB-1.0 Safety Guide (42)

The design of the electrical systems for the nuclear installation shall be sufficiently resistant against failure of installation important to nuclear safety as a result of a single failure and against their common-cause failures. In order to achieve that, the principles of physical separation, functional isolation and independence, redundancy, and physical and functional diversity shall be sufficiently applied.

References: Decree No. 195/1999 Coll., Article 7, SÚJB BN-JB-1.0 Safety Guide (43), IAEA SSR 2/1 Req. 30, 5.48

The electrical systems and their components shall also be designed to cope with the surrounding environment's design properties, including EMC. Electrical systems shall be sufficiently dimensioned in respect of general technical properties (e.g. output, voltage, electrical and mechanical short circuit resistance).

References: SÚJB BN-JB-1.0 Safety Guide (20)

The electrical systems shall meet the requirements of the basic conceptual Czech national standards or the supplier's equivalent norms and also the Czech legislature's requirements relating to occupational health and safety. The utilized technical codes, standards, requirements, rules, and computation programs used for electrical system designing shall be clearly specified, and their suitability for nuclear facilities shall be secured and verified and they shall be in line with the internationally accepted practice. If their combination is used, it shall form a consistent whole, and the mutual compatibility and usability of its individual components shall be proven. Foreign documents and programmes must guarantee such a level of justified interest protection that is identical to the one in the Czech Republic.

3.8.1.1.1 Provision of auxiliary power supply

References: Decree No. 195/1999 Coll., Article 29 (2); SÚJB BN-JB-1.0 Safety Guide (103)

The electrical installation feeding installation important to nuclear safety shall provide, besides normal and stand-by feeding, emergency source feeding, i.e. the power supply of this installation shall be backed up through adequate power capacity available for a necessary period in order to secure reliable systems operation regardless of the availability of their own generators or the external power supply system. The information and control systems, the most critical information systems, emergency lighting, and other necessary installation shall be powered by uninterruptible power sources.

References: Decree No. 95/199 Coll., Article 3, SÚJB BN-JB-1.0 Safety Guide (10)

The design solution shall meet the requirements of all the necessary systems and installation requiring uninterruptible power supply due to safety reasons. Due to the defence in depth principle application, an adequate level of backed up power supply shall also be provided to systems associated with safety or investment protection related systems.

3.8.1.1.2 Power supply system backup

References: Decree No. 195/1999 Coll., Article 30 (1)

Systems that shall be backed up due to nuclear safety reasons shall be supplied with power in a way that shall guarantee their functional independence, i.e. power supply systems and their sources of power, including emergency power supply systems and sources, shall be mutually independent. If the number of power sources is smaller than the number of independent systems, the design shall prove that such a fact does not decrease their reliability.

References: Decree No. 195/1999 Coll., Article 3, SÚJB BN-JB-1.0 Safety Guide (10)

The design solution shall meet the requirements of all the necessary systems and installation requiring uninterrupted power supply due to safety reasons. Due to the defence in depth principle application, adequate level of backed up power supply shall also be provided to systems associated with safety or investment protection related systems.

References: Decree No. 195/1999 Coll., Article 30 (3)

If some system's flawless operation is critical for nuclear safety, the power supply system shall provide necessary power without any limitations even during simple failure.

References: Decree No. 195/1999 Coll., Article 30 (2)

If a simple failure of power supplied systems does not impact their function, a simple failure of the power system or its source is acceptable as well.

3.8.1.1.3 Emergency power sources

References: Decree No. 195/1999 Coll., Article 31 (1)

Systems requiring uninterrupted power supply (category I appliances) shall be powered via sources providing power immediately (batteries or batteries with inverters or converters).

References: Decree No. 195/1999 Coll., Article 31 (2)

Power sources and power supply systems only activated after a particular period of abnormal lost of power supply or emergency conditions (category II backed up power supply grids) or other technical means meeting the safety function requirements shall achieve the necessary operating parameters quickly enough, compared to the outermost required period determined through a safety analysis/design that is necessary to activate the powered appliances.

References: SÚJB BN-JB-1.0 Safety Guide (104)

Emergency power supply systems shall be able to supply installation important to nuclear safety at the adequate level in all operating states, as well as during basic design basis accidents, even in the case of their simple failure and concurrent loss of external power supply. If the number of emergency power supply sources is smaller than the number of independent systems, the design must prove that their reliability and functional independence shall not be limited.

References: Decree No. 195/1999 Coll., Article 31 (3), SÚJB BN-JB-1.0 Safety Guide (49)

It shall be possible to subject the emergency power supply systems to functional tests.

3.8.1.2 EXTERNAL ELECTRICAL SYSTEM

The external electrical system leads the output to the electrification system and it also secures its own normal and stand-by power supply.

The external electrical system represents a crucial interface between the power plant and the electrification system. This interface impacts the operation of systems important to nuclear safety and it also plays an important role in the defence in depth hierarchy in relation to power supply.

The external electrical system of each ETE3,4 unit includes electrification system connection (through the 400 kV and 110 kV Kočín switching station), the 400 kV output line, the 110 kV stand-by power supply line, generator transformer or transformers, own generator or generators and also auxiliary normal and stand-by transformers, including adequate accessories of the specified systems.ETE3,4

3.8.1.2.1 The requirements for connecting ETE3,4 to the electrification system

References: Decree No. 195/1999 Coll., Article 29 (1)

The output outlet of every ETE3,4 unit shall make sure as follows:

- External and internal failures of the power distribution grid impact the reactor's operation and the heat transmission systems as little as possible
- The operation critical systems can be powered from two different electric power sources (own turbo-generator and the electrification system grid)

References: SÚJB BN-JB-1.0 Safety Guide (98), IAEA SSR 2/1 Req. 41

The ETE3,4 design shall include potential interactions between the new ETE3,4 units and the current ETE1,2 units and the electrification system grid, including the independence and the number of output outlets and stand-by power supply. The systems and components securing the connection between ETE3,4 and the grid shall achieve the necessary reliability required for the powering of the power plant's systems important to safety.ETE3,4ETE3,4ETE3,4

References: IAEA SSR 2/1, Req. 41

The functionality of systems important to nuclear safety shall not be negatively affected by electric grid failures, including any expected supply grid voltage and frequency fluctuations.

3.8.1.2.2 The requirements for the external electrical system

References: Decree No. 195/1999 Coll., Article 29, Safety Guide SÚJB BN-JB-1.0 (98), (100)

The design solution of the output outlet of every ETE3,4 unit and its stand-by power supply shall adequately make them resistant to failures originating outside ETE3,4 and inside ETE3,4 and the solution shall also make sure it does not generate failures that could negatively impact its neighbouring NPP units of the electrification system grid. It means that the normal power supply (PNVS) and stand-by power supply (RNVS) systems of every ETE3,4 unit shall be designed in such a way that during no simple failures of the connection between own power supply sources and the electrification system grid shall simultaneous power supply failures of the PNVS and RNVS sources of a given unit be caused.ETE3,4ETE3,4ETE3,4ETE3,4

References: Decree No. 195/1999 Coll., Article 3; SÚJB BN-JB-1.0 Safety Guide (10), (98), (99), (100)

The stand-by auxiliary power supply source of every ETE3,4 unit shall be able to sufficiently replace the normal power supply sources during the unit's normal and abnormal operation and also under emergency conditions affecting the unit.ETE3,4 This capability must also be secured during simple failures of systems or components and scheduled inspection periods of the internal electrical system (ETE3,4) and the external electrical system (adequate part of the electrification system).

3.8.1.3 THE ELECTRICAL SYSTEMS INSIDE THE POWER PLANT

The internal electrical system provides the normal, stand-by, and emergency auxiliary power supply, including the systems important to nuclear safety. The main task of the internal electrical system is to secure the required qualitative and quantitative power supply parameters that shall enable reliable operation of the powered technological systems (i.e. mainly reliable electric power generation), and in the case of systems important to nuclear safety, it shall also enable reliable fulfillment of all required safety functions.

The electrical systems inside the ETE3,4 power plant include all the electrical systems and components that are not part of the external electrical system defined in section 3.8.1.2.

3.8.1.3.1 The general requirements for the electrical systems inside the power plant

References: Decree No. 195/1999 Coll., Article 3, SÚJB BN-JB-1.0 Safety Guide (10), IAEA SSR 2/1 Req. 7

To limit the impacts of the failures of power sources feeding auxiliary power supply of ETE3,4, the defence in depth principle shall also be prepared for and applied to the electrical systems. It shall also include, among others, automatic sequential and controlled transition of electric power supply sources from normal power supply (PNVS) to stand-by power supply (RNVS) or to emergency power sources.ETE3,4

References: SÚJB BN-JB-1.0 Safety Guide (6), (7), IAEA SSR 2/1, Req. 22

The electrical systems shall meet the requirements specified in the ETE3,4 design concerning the power supply of systems/appliances, especially machinery-technological installation, constructions, and the instrumentation and control systems. These requirements for electrical systems shall derive from the basic classification of systems, components, installation and structures into important and unimportant ones in terms of nuclear safety and the categorization of selected installation into safety classes (SC) in line with fulfilled safety functions (see also sections "3.3.2.1 The basic legislative requirements for the design of structures, components, systems, and installation" and "3.3.2.2 The classification of structures, components, and systems in the tentative design concept"). This safety classification shall form the basis for the determination of specific requirements for the design solution robustness, the qualification and reliability of electrical systems, including the requirements for the maximum acceptable period of power supply interruption, sufficient capacity, output, voltage, and other parameters.

References: SÚJB BN-JB-1.0 Safety Guide (98), (42)

The electric power supply systems shall be designed in such a way that both external and internal failures of the distribution system shall minimally affect the operation of ETE3,4. That is why the powering of auxiliary power supply appliances shall be divided among a few distribution facilities, power supply systems, and power sources, and built in such a way that external and internal failures of the electrical systems shall minimally impact the reactor's operation, heat removal systems, and also power generation systems.

References: SÚJB BN-JB-1.0 Safety Guide (42), IAEA SSR 2/1, Req. 27, 5.42

The durability and robustness of the scheme, the level of reliability, redundancy, diversity and independence of the electrical systems shall be adequate to the requirements for robustness, reliability, and other functional requirements of powered systems/appliances that generally derive from the requirements of the technological systems of ETE3,4 (especially the mechanical-technical, constructions, and the instrumentation and control systems) and to their importance in terms of nuclear safety.

3.8.1.3.2 The requirements for the regular power supply electrical systems

References: Decree No. 195/1999 Coll., Article 3; SÚJB BN-JB-1.0 Safety Guide (10), (100), (102)

Auxiliary power supply of every ETE3,4 unit designed to feed appliances not requiring backup power supply from emergency electric power supply sources (normal power supply grid) shall be mainly powered from its own turbo-generator or the 400 kV network from the unit's output outlet system, i.e. the operating power supply feeding its auxiliary power supply (PNVS). In the case of a PNVS failure, auxiliary power supply shall be fed from the stand-by power supply grid, i.e. the stand-by auxiliary power supply source (RNVS).ETE3,4

References: Decree No. 195/1999 Coll., Article 3, SÚJB BN-JB-1.0 Safety Guide (100)

The design solution shall include measures reasonably minimizing the probability of simultaneous failures of the normal and stand-by power sources during normal operation or during failures expected by the design or caused by the surrounding environmental conditions.

References: SÚJB BN-JB-1.0 Safety Guide (98), (99)

The normal power supply sources designed for auxiliary power supply shall be able to secure power supply during the unit's normal operation, abnormal operation, and also under emergency conditions, provided such power supply is available.

References: Decree No. 195/1999 Coll., Article 3, SÚJB BN-JB-1.0 Safety Guide (10), (99)

The stand-by power supply sources designed for auxiliary power supply shall be able to sufficiently replace the normal power sources feeding auxiliary power supply during the unit's normal operation, abnormal operation, and also under emergency conditions, provided such power supply is available (especially in relation to power and short-circuit conditions).

References: Decree No. 195/1999 Coll., Article 3, SÚJB BN-JB-1.0 Safety Guide (10)

The stand-by power sources of both ETE3,4 units shall be able to partially back up each other.ETE3,4

3.8.1.3.3 The requirements for the backup power supply electrical systems

References: Decree No. 195/1999 Coll., Article 3, SÚJB BN-JB-1.0 Safety Guide (10)

In the case of simultaneous failures of auxiliary power supplies of every ETE3,4 unit based on the normal and stand-by power sources, systems importatnt to nuclear safety shall be powered by emergency power sources via backup power supply systems (SZN).ETE3,4

References: Decree No. 95/199 Coll., Article 30, SÚJB safety Guide BN-JB-1.0 (10), (42)

The backup power supply systems and their emergency sources shall be independent of the unit's normal and stand-by auxiliary power supply sources.

References: SÚJB BN-JB-1.0 Safety Guide (42), (43), IAEA SSR 2/1, Req. 68, 6.44

The backup power supply systems and their emergency power sources shall be designed in such a way that they shall fulfill the requirements of fed appliances of systems important to safety (robustness, reliability, mutual independence, redundancy, capacity, power supply quality, the backup power supply systems' emergency power sources switching period, resistance to environmental impacts, including EMC, and seismicity). The back-up power supply systems intended for safety systems' power supply shall form a power supply system resistant against a simple failure and against common-cause failures within each unit.

References: Decree No. 195/1999 Coll., Article 4 (2), SÚJB BN-JB-1.0 Safety Guide (49), IAEA SSR 2/1 Req. 29

The back-up power supply systems and their emergency power sources shall also be designed to enable checking of their state, testing of their functional capabilities and reliability, inspection and repair activities, component replacements, and calibrations without any negative impacts on nuclear safety.

References: Decree No. 195/1999 Coll., Article 31 (2), SÚJB BN-JB-1.0 Safety Guide (105)

Emergency sources of category II of backed up power supply and their control systems shall be able to undertake power supply of the back-up power supply system appliances, in accordance with the programme of sequential loading within a period of time that is shorter than required for their power supply. It shall further enable subsequent connection and disconnection of appliances, operator or automatic interventions, and shall also provide power supply of appliances for the technologically needed time. The design shall consider various combinations and time sequences of technological failures and loss of power supply.

Note:
Requirements for category II emergency power sources for backed up power supply and relevant control systems shall be applied provided that they exist in the design. Vice-versa, they shall not be relevant in case of a different design solution, i.e. when different technical means than category II emergency power sources are used, if such systems comply with all requirements for safety functions.

References: Decree No. 195/1999 Coll., Article 4 (1); SÚJB BN-JB-1.0 Safety Guide (105)

In the course of the design's load-applying sequence and during the operation of emergency power sources, there shall occur neither overload nor shutdown as a result of intervention of protections, limiters, or instrumentation and control systems. Logics of the control automatics and systems shall prevent repeated initiation of a preparatory and load-applying sequences during transients in the power supply grid if the back-up power supply system is already fed from the category II emergency power source of the backed up power supply.

References: SÚJB BN-JB-1.0 Safety Guide (101)

Auxiliary power supply systems shall be equipped with monitoring systems that will provide the respective operator with detailed information on operation states and important electrical parameters of monitored emergency power sources, connected safety and auxiliary systems, and the system controlling their start-up to their functions, in accordance with protective system requirements. The monitoring system shall operatively detect and localize a malfunction (if any) of the monitoring systems and their components.

3.8.1.4 STATION BLACKOUT

A Station Blackout (SBO) event is one of the postulated initiation events that can initiate occurrence of accident conditions, including severe accidents (see Section "3.19 Probability analyses and assessment of design extension conditions").

The Station Blackout event refers to concurrent loss of normal and stand-by auxiliary power supply of both ETE3,4 units, i.e. loss of power supply from the external electrical system (400 kV, as well as 110 kV grids)ETE3,4), whereas in addition to that, generator shutdown occurs for at least one of the units. The unit with a shut down generator will also experience failure of emergency sources of AC power supply for category II backed up power supply systems concurrently in all safety system divisions, i.e. only category I systems' emergency sources of power supply will remain serviceable. In the adjacent unit of the electrical station, at least one safety system division will remain serviceable, including emergency sources of AC power supply for category II backed up power supply systems.

References: SÚJB BN-JB-1.0 Safety Guide (36), (37)

A possibility of occurrence of the full loss of ETE3,4 power supply shall be examined and considered by the design, covering a concurrent loss of all external power supply sources and shutdown of a turbo-generator, including a loss of all emergency power supply sources for backed up power supply for category II appliances (Station Blackout event) if such sources exist in the design.

References: Decree No. 195/1999 Coll., Article 3, SÚJB BN-JB-1.0 Safety Guide (10)

Based on the analysis of a possibility of occurrence of the full loss of power supply and in accordance with the application of the defence in depth principle, the design shall assess installation of an alternative AC power supply source (AAC) in order to manage and recover from SBO event, by means of preventive measures against the event's evolving in the electrical systems into a subsequent severe accident in the unit's nuclear-technical part. The AAC source shall be independent of normal power supply (PNVS) and stand-by power supply (RNVS), as well as of the emergency sources of category II back-up power supply if they exist in the design, whereas unfavourable consequences of a SBO event must be minimized for the AAC source.

References: Decree No. 195/1999 Coll., Article 4 (1); SÚJB BN-JB-1.0 Safety Guide (43)

For systems and components dedicated to managing SBO event, the design shall prove their ability to fulfill required functions under SBO conditions.

3.8.2 PROPERTIES OF THE ELECTRICAL SYSTEM DESIGN FOR THE PURPOSE OF THE PRELIMINARY ASSESSMENT

This section describes the properties of the design for the needs of a preliminary assessment of the design concept. The information for the specification of the design properties was based on the technical part of the tender documentation stipulating the requirements for safety and technical design of the future power plant design. This section includes partial assessment as to whether the tentative concept of the design segment in question complies with legislative requirements specified in Sections 3.8.1.1 through 3.8.1.4. The scope of this section includes identification and assessment of the general level of the requirements for the functions of the electrical systems, whereas particulars of the specific technological implementation shall be included in the design documentation of the selected supplier of the NPP, and the assessment will be performed within the next stage of the safety documentation.

The electrical system design shall be designed so as to ensure meeting of the basic safety requirements and principles described in Section 3.3.1. These include, in particular, the principle of defence in depth and the design criteria specified for the systems, structures, and components within the design basis, including the relevant design requirements based on the conditions of the site, as specified in Chapter 2 and summarized for the most critical ones in Section 2.10. Legislative requirements derived from specified legislative supporting documents for processing of ISAR are summarized in Sections 3.8.1.1 through 3.8.1.4.

The design shall include the following electrical systems and related subsystems, whereas the particular method of their implementation will be specified in the draft of the particular design solution of the nuclear power plant. The design solution selected for implementation does not have to include all systems, structures, and components specified in this Section; however, if the selected design utilizes them, they shall fulfill applicable requirements specified in this Section.

3.8.2.1 GENERAL REQUIREMENTS FOR ELECTRICAL SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

The electrical systems shall be designed as a block unit so that the possibility of transfer and spread of electrical failures between individual units is sufficiently limited. Each ETE3,4 unit shall have available grids and sources specified in more detail within this section of ISAR. The electrical systems shall be designed in accordance with requirements of mechanical-technical systems, instrumentation and control systems, construction systems, and in accordance with specified requirements for fulfillment of specified safety functions in such a way that the specified level of fulfillment of the defence in depth principle, the resistance against common-cause failure principle, and fulfillment of a simple failure criterion would not decrease.ETE3,4

Auxiliary power supply grids

Power supply of appliances important to nuclear safety shall be realized from category I and II grids of backed up power supply in terms of the Czech legislation regarding safety systems ("Emergency power supply sources for safety systems", see below in this Section). Power for such grids shall be regularly supplied from the normal auxiliary power supply (PNVS) or the stand-by auxiliary power supply (RNVS), whereas in case of concurrent failure of PNVS and RNVS, they shall automatically switch to power supply from emergency sources. Sources and grids of category I and II backed up power supply shall be arranged into backedup power supply systems. Such backed up power supply systems shall be sufficiently mutually independent, separated (from the electrical, construction, fire-prevention, etc. aspects), and resistant to impacts of the ETE3,4 environment, including provision of the necessary level of electrical-magnetic compatibility (EMC) and seismicity. They shall also be resistant to transients generated by the external electrical system (i.e. the superior electrification system) and the internal electrical system (i.e. the auxuiliary power supply installations). The backed up power supply systems shall form supporting safety systems for a given division of operational and protective safety systems.ETE3,4

The normal power supply grids (i.e. category III of non-backed-up power supply) shall only be powered from normal and stand-by sources, i.e. without possibility to apply emergency power source backups. These grids shall power appliances not important to nuclear safety. The normal power supply grids shall reasonably back each other up and they shall also feature the necessary EMC level and resistance to transients generated by the external and internal electrical systems.

The normal electric power sources

The normal power sources shall be an integral part of the hierarchy of the defence in depth principle in relation to the power supply of ETE3,4. Auxiliary normal transformers with on-load voltage regulation shall be the normal source of auxiliary power supply of every unit. These auxiliary normal transformers shall be directly powered from the given unit's alternator (if the branch is installed between the generator's terminals and the generator transformer) or from a location behind the generator transformer on the 400 kV side (if the branch is installed on the extra high voltage side of the generator transformer in the power plant's 400 kV switching station on the NPP's area). This type of power supply shall be based on generator circuit breakers and unit circuit breakers. As a result, auxiliary power supply shall be fed from the normal sources in all regimes when the function and the connection to the generator or the external electrical system shall be maintained (starting, normal operation at the required output, regular shutdown, abnormal regimes, and emergency conditions in the unit, transmission system or turbo-machinery failures with maintained stability of the turbo-machinery or the grid.ETE3,4

The stand-by electric power sources

The stand-by sources shall be an integral part of the hierarchy of the defence in depth principle in relation to the power supply of ETE3,4. An auxiliary stand-by transformer (possibly transformers) with on-load voltage regulation shall be the stand-by source for every unit. The transformer shall be connected to the 110 kV distribution grid. The unit shall switch to stand-by power supply upon partial or total loss of its normal power sources. Switching shall be initiated preferably automatically, but manual switching shall be possible as well. In connection with the application of the defence in depth principle to the electrical systems, stand-by power supply shall partially compensate for operational power supply as necessary in order to avoid emergency power supply sources activation.ETE3,4

The emergency electric power sources for the systems associated with safety and investment protection

The emergency power sources shall be integrated into electrical systems based on the hierarchy of the defence in depth principle in relation to the power supply of ETE3,4. These power sources shall feed systems classified as systems associated with safety, costly systems, mostly conventional islands (for example, a turbo-machinery), and other installations important to personal safety.

The emergency electric power sources for the safety systems

The emergency power sources for the safety systems shall fulfill safety functions, and they also represent a part of the hierarchy of the defence in depth principle in the power supply of ETE3,4.

These power sources shall be designed for the case of simultaneous loss of both normal and stand-by power sources and they shall be structured into backed up power supply systems. These backed up power supply systems, including the emergency power supply sources for safety systems (category I and II of backed up power supply) shall be strictly designed on the block unit basis.

The emergency power sources shall be installed on the ETE3,4 area and their operation shall not be impacted by the status of both normal and stand-by power sources and the status of the external electrical system.ETE3,4

The emergency power sources for the category II backed up power supply grids shall comprise diesel generators or other technical means meeting the safety function requirements. In case of using diesel generators, their start shall be initiated automatically when power supply of the respective backed up power source from normal power supply (PNVS) or stand-by power supply (RNVS) is lost, and their load shall be applied automatically in a gradual (sequential manner) after their connection to the respective backed up power source. The redundant channels of emergency sources initiation shall sufficiently feature the principle of functional, design, and technological diversity.

The emergency power sources for the category I backed up power supply grids shall comprise accumulators with uninterruptible power supply networks shall comprise accumulators with uninterrupted power supply systems (rectifiers and inverters). The design of category I backed up power sources shall, if necessary, feature the principle of functional, design, and technological diversity.

Alternative AC power sources

The alternative AC power sources shall be designed to recover after Station Blackout and similar events. The basic legislative requirements relating to Station Blackout are specified in section "3.8.1.4 Station Blackout" and the associated design concept assessment is available in section "3.8.2.4 Solving Station Blackout events in the tentative design concept".

Partial preliminary assessment

The tentative design concept summarizing the most critical requirements for systems, structures, components, and the safety and technological functions of electrical systems specified in section "3.8.2.1 General requirements for electrical systems in the tentative design concept" creates conditions for meeting the requirements of Decree No. 195/1999 Coll. [L. 266] Article 3, Article 7, Article 29 (2), Article 30 (1)(2)(3), Article 31(1)(2)(3), document IAEA SSR 2/1 [L. 252] Req. 30, 5.48, and SÚJB Safety Guide BN-JB-1.0 [L. 276] (10, 20, 42, 43, 49, 99, 103, 104, and 106).

3.8.2.2 THE EXTERNAL ELECTRICAL SYSTEM IN THE TENTATIVE DESIGN CONCEPT

The external electrical system design shall secure such a solution of the unit's connection to the electrification system that shall meet the technological and safety requirements of the internal electrical system specified in section "3.8.1.3 The electrical systems inside the power plant" (i.e. the auxiliary power supply system).

The design of the external electrical system and its connections to the internal electrical system shall feature the defence in depth principle. It means that in the case of electric failure initiated in the external electrical system the required sequence of transitioning the auxiliary power supply from normal power supply to stand-by power supply and, if necessary, to emergency power supply, shall be secured. The defence in depth principle applied to the external electrical system shall be technically fulfilled through the adequate level of independence, separation, functional isolation, redundancy, and physical and functional variety (diversity).

The ETE3,4 units shall be connected to the electrification system of the Czech Republic through the 400 kV Kočín switching station to which the ETE1,2 units are currently connected. Both ETE3,4 units shall be connected to the 400 kV Kočín switching station via 400 kV and 110 kV external transmission lines leading through the same corridor. Every unit shall be connected to its own 400 kV output outlet line and one independent or sufficiently technologically independent 110 kV stand-by power supply line.

The process subsystems for power outlet shall be designed for each unit so that a possibility of transfer and spread of electrical failures between the individual units is sufficiently limited. Upon transformation in generator transformers, the output of every generator or generators of a given unit is led to the 400 kV transmission system through a unit line. A generator circuit breaker shall be installed in the generator outlet or behind the generator transformer. An extension of the output outlet system (generator voltage or the 400 kV branch behind the generator transformer) shall provide operational power supply for the unit's auxiliary power supply. The unit initiation shall be provided from the normal power supply, i.e. the 400 kV line system. In order to provide high reliability, "two circuit breakers for one branch " shall be installed for unit power outlet. An unit breaker for the 400 kV line shall be installed in the Kočín substation. The power output electrical systems of each unit shall be equipped with suitable protections, automatics, regulations, and control systems coordinated with corresponding technical systems of the 400kV line and all relevant electrical subsystems within the auxiliary power supply diagram for ETE3,4. For connection of ETE3,4 units, the 400kV substation shall be expanded, modified, and reinforced, and it shall be one of the normal power sources for auxiliary power supply.ETE3,4ETE3,4

The standby power supply shall also be designed for each unit and shall provide the required level of functional independency of auxiliary power supply. The 110 kV Kočín substation, which is also a standby power source for auxiliary power supply of the ETE1,2 units, shall be the standby power supply for auxiliary power supply of both ETE3,4 units. The existing 110 kV Kočín substation is interconnected with 400 kV transmissions system by 400/110 kV transformation and has also a direct link to the 110kV Dasný substation. Capacity of the 110kV substation, as well as the 400/110kV transformation in the Kočín substation shall be reinforced and modified for this purposes. The electrical systems of each unit's standby power supply shall be equipped with suitable protections, automatics, regulations, and control systems coordinated with corresponding technical systems of the 110kV distribution system and with the power supply diagram for the auxiliary power supply of ETE3,4.ETE3,4ETE3,4

Partial preliminary assessment

The tentative design concept summarizing the major requirements for the systems, structures, and components and on the safety and technological functions of the electrical systems described in Section "3.8.2.2, External electrical system in the tentative design concept", creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 29 (1), IAEA SSR 2/1 [L. 252] Req. 41 and SÚJB BN-JB-1.0 Safety Guide [L. 276] (10, 98, 99, 100).

3.8.2.3 ELECTRICAL SYSTEMS INSIDE THE POWER PLANT IN THE TENTATIVE DESIGN CONCEPT

The defence in depth principle shall feature in the design of the internal electrical system and its connections to the external electrical system. It means that in the case of electrical failure initiated in the internal electrical system, the required sequence of transitioning the auxiliary power supply from normal power supply to stand-by power supply and, if necessary, to emergency power supply, shall be secured. That means that in case of failure of the normal power supply, the automatic transition of auxiliary power supply to the stand-by power supply shall be secured, and in case of concurrent failure of the normal and stand-by power supply, there shall be emergency power supply sources available, and they shall include at least three backed up power supply systems independent as for their functioning and dispositions, whereas each of such backed up power supply sources shall be sufficient for fulfillment of required safety functions.

Implementation of the defence in depth principle shall be technically fulfilled through the adequate level of independence, separation, functional isolation, redundancy, and physical and functional variety (diversity) of safety systems, and possibly also systems related to safety if they fulfill safety functions and alsosystems associated with investment protection.

The internal electrical system shall be designed as a block unit so that a possibility of transfer and spread of electrical failures between individual units is sufficiently limited.

The internal electrical system shall fulfill the general requirements for electrical systems stipulated in Section "3.8.2.1 General requirements for electrical systems in the tentative design concept".

Partial preliminary assessment

The tentative design concept summarizing the most critical requirements for systems, structures, components, and the safety and technological functions of electrical systems specified in section "3.8.2.3 Electrical systems inside the power plant in the tentative design concept" creates preconditions for meeting the requirements of Decree No. 195/1999 Coll. [L. 266] Article 3, Article 4 (1)(2), Article 31 (2), of document IAEA SSR 2/1 [L. 252] Req. 7, 22, 27 (5.42), 29, 68 (6.44) and SÚJB BN-JB-1.0 Safety Guide [L. 276] (6, 7, 10, 42, 49, 98, 99, 100, 101, 102, 105).

3.8.2.4 SOLUTION OF A STATION BLACKOUT EVENT IN THE TENTATIVE DESIGN CONCEPT

In the electrical system design, an analysis of occurrence of the total loss of power supply shall be conducted, and based on it, installation of alternative alternate power supply sources (AAC), which will be especially intended to manage and recover from the events of the Station Blackout (SBO) type, shall be assessed. The electrical systems for SBO shall form a part of the defence in depth system in the area of ensuring power supply for ETE3,4, and they shall be especially intended for managing of accident conditions, including severe accidents.

The alternative AC sources for SBO shall be designed and dimensioned so that required safety functions are provided in the requested scope. These systems' design solution and their integration into the electrical systems shall sufficiently apply principles of physical separation, functional isolation, redundancy, and physical and functional diversity.

Partial preliminary assessment

The tentative design concept summarizing the most critical requirements for systems, structures, components, and the safety and technological functions of electrical systems specified in section "3.8.2.4 Solution of Station Blackout events in the tentative design concept" creates preconditions for meeting the requirements of Decree No. 195/1999 Coll. [L. 266] Article 3, Article 4 (1), and SÚJB BN-JB-1.0 Safety Guide [L. 276] (10, 36, 37, 43).

3.8.3 COMPREHENSIVE PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT

The ETE3,4 design shall meet the safety requirements specified in Section "3.3.1.1.1 Basic legislative requirements for provision of safety", as well as the requirements specified in Sections 3.8.1.1 through 3.8.1.4, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and related implementing decrees regarding nuclear safety, radiation protection, and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270] documents.

To meet those requirements, the ETE3,4 design shall use systems, structures, and components and their equipment integrated within electrical systems, which shall ensure (with adequate reliability and resistance) qualitatively and quantitatively adequate safety and technological functions, in accordance with the specified requirements for the prescribed safety functions.

The principles of the design solution described in Section "3.8.2 Description of the design for the purpose of preliminary assessment" were set up based on the requirements that the applicant for the licence imposed on the potential suppliers of the nuclear installation within the tender, and they make up the concept of the design solution for this part of the design. The partial assessments performed proved that the expected design of the electrical systems creates preconditions for compliance with the relevant requirements for systems, structures, and components and safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], SÚJB BN-JB-1.0 Safety Guide [L. 276], and the IAEA SSR 2/1 [L. 252], and WENRA [L. 27] documents.

The particular method of technical implementation of individual requirements for operation of the electrical systems specified in Section 3.8.2 shall be specified in detail only in the design documentation of the selected nuclear power plant supplier.

3.9 SUPPORTING SYSTEMS

This comprehensive part of the Initial Safety Analysis Report is structured into three basic sections:

The opening Section 3.9.1 summarizes, analyses, and specifies legislative requirements for supporting systems, with focus on systems handling fuel and its storage, water systems, and selected operation supporting systems, such as the compressed air system, sampling systems, and the coolant refilling and treatment system. This Section also includes basic requirements for heating, ventilation, and air conditioning systems, lighting systems, and diesel generator supporting systems, and also requirements for the fire-prevention programme.

Section 3.9.2 that follows contains description and specification of the basic requirements for the functions of the supporting systems specified in the opening Section 3.9.1 in a form that summarizes the applied design requirements for these systems in relation to all the relevant designs involved in the current tender procedure. The section's goal is to formulate the design's general properties for the purposes of partial preliminary assessment.

The final Section 3.9.3 contains a comprehensive preliminary assessment of the concept of the supporting systems, summarizing the conclusions of the partial preliminary assessments presented in Section 3.9.2. The assessment of the summarized requirements contains a tentative design concept assessment required by the law.

At the next stage of the licensing documentation, the applicant shall provide evaluating and supporting information on the selected design that will make it possible to assess the ability of the supporting systems to support superior systems during fulfillment of the specified safety functions during the whole lifetime of the nuclear unit in any defined operational state and in accident conditions. The section shall also be supplemented with detailed information at the depth and in the structure described in RG 1.206 [L. 275]. This Section shall contain identification of supporting systems important to safe unit shutdown or important for maintaining health and safety of residents. For every system, there shall be design bases for the most critical components, description, safety evaluation, reliability evaluation, and a scope of controls and tests.

3.9.1 BASIC LEGISLATIVE REQUIREMENTS FOR THE SUPPORTING SYSTEMS

3.9.1.1 FUEL HANDLING AND STORAGE

This section describes basic legislative requirements for the design of the equipments for handling of fresh, irradiated, and spent fuel and for its storage.

The fuel handling and storage system's installation serves the purpose of receipt and storage of fresh fuel, filling and removing fuel from a reactor, storage of irradiated fuel in pools, and for handling irradiated fuel and packaging assemblies. Among its other important functions are creation of fuel reserve for refuelling, protection of stored fuel from mechanical damage and pollution, and maintenance of the sub-critical state of fuel in all storage and handling modes, even under the most unfavourable conditions affecting reactivity, even in case of seismic activities.

3.9.1.1.1 Fresh fuel handling and storage

References: Decree No. 195/1999 Coll., Article 46, SÚJB BN-JB-1.0 Safety Guide (121)

The fresh fuel handling and storage installation shall be designed so that:

- With a margin, it prevents achievement of the critical state, even under the conditions of the most efficient deceleration of neutrons (optimal moderation) by means of a spatial layout or other physical means and procedures, which would prevent:
    - Exceeding of the value of 0.95 of the effective coefficient of neutron multiplication under assumed accident situations derived from a design failure (including flooding)
    - Exceeding of the value of 0.98 of the effective coefficient of neutron multiplication under the optimum moderation conditions
- It enables prevention of periodical inspections and tests
- It decreases the possibility of fuel damage or loss to the minimum
- It enables safe handling of fresh nuclear fuel and prevents falls during fuel transport
- It prevents falling of heavy items on the fuel assembly, i.e. items whose weight is greater than the weight of the fuel assembly
- It enables storage of fuel elements or assemblies for structures and operating units containing a reactor

3.9.1.1.2 Irradiated and spent fuel handling and storage

References: Decree No. 195/1999 Coll., Article 47; SÚJB BN-JB-1.0 Safety Guide (121); IAEA SSR 2/1, Req. 80, 6.64, 6.65, 6.66, 6.67, 6.68

Installation for irradiated and spent fuel handling and storage, as well as handling and storage of other materials containing radionuclides, shall form a part of ETE3,4. They shall be protected by physical barriers and defence in depth ensured especially by means of inherent properties of the systems or by passive means.

The irradiated and spent fuel handling and storage installation shall be designed so that:

- With a margin, it prevents achievement of the critical state, even under the conditions of the most efficient deceleration of neutrons (optimal moderation) by means of a spatial layout or other physical means and procedures, which would prevent:
    - Exceeding of the value of 0.95 of the effective coefficient of neutron multiplication under assumed accident situations (including flooding)
    - Exceeding of the value of 0.98 of the effective coefficient of neutron multiplication under the optimum moderation conditions

- It ensures sufficient residual heat removal in normal and abnormal operation, as well as under accident conditions of a design failure
- It ensures a possibility to carry out periodical inspections and tests, especially from the aspect of integrity monitoring of fuel elements and assemblies
- It prevents falling of irradiated fuel during transport
- It decreases the possibility of fuel damage to the minimum, i.e. it especially prevents subjecting the fuel element or fuel assembly to impermissible exertion during handling
- It prevents falling of heavy items on the fuel assembly, i.e. items whose weight is greater than the weight of the fuel assembly
- It enables storage of damaged fuel elements or assemblies for structures and operating units containing a reactor and their handling even when they cannot be handled by standard design means
- It enables radiation protection of the nuclear installation's employees, the population, and the environment
- It provides for wet storages with water filling
    - Check of the chemical composition and content of radionuclides in all water in which irradiated fuel is stored or in which it is handled
    - Monitoring and management of the water level height in a spent fuel pool and detection of leaks
- It ensures the possibility of hermetically sealing spent nuclear fuel that does not comply with respective design criteria regarding the integrity of spent fuel or other materials containing radionuclides
- It has the design capacity, which would enable rearrangement of fuel assemblies and/or packaging assemblies for the purpose of inspection, damage finding, and repairs
- After termination of ETE3,4 operation and during the shutdown, or in the case of unforeseen operation problems, it enables safe handling with packaging assemblies with spent nuclear fuel or other materials containing radionuclides if they are used or with individual fuel assemblies during the time period specified in advance
- It prevents uncovering of fuel assemblies in the pool if piping breaks, i.e. measures against spontaneous drainage of the pool.

3.9.1.2 WATER SYSTEMS

This section describes basic legislative requirements for the design of safety-critical water systems.

Water systems provide heat removal from important appliances of the production unit and their reliable operation and cooling of technical equipments in the reactor building that come into contact with radioactive materials.In addition, they provide additional water for adding into the secondary and primary circuits, utility and drinking water, storage of condensate, handling of non-active waste water, including seepage, cooling water for cooling appliances in the machine room and in reactor building of the individual units, including fire water and a facility for additional water treatment.

Safety-critical water systems that are closely related to the nuclear safety enable reliable shutdown of technical equipments and cool down using cooling water in all operating modes of the unit, as well as under accident conditions.

3.9.1.2.1 Nuclear safety-critical water systems

Designs of water systems and related technical circuits shall comply with the following requirements:

References: Decree No. 195/1999 Coll., Article 8(1)

The technological assemblies and installation involved in the removal of heat released by nuclear fission, residual heat, and operational heat, shall reliably ensure adequate reactor cooling during both normal and abnormal operation as well as in accident conditions included in the design bases.

References: Decree No. 195/1999 Coll., Article 25 (2), Article 8 (2)

The residual heat removal system shall ensure adequate redundancy of its important equipments, suitable interconnection, possibility to disconnect parts of the system, and detection of leaks and possibility of their capture, so that the system will work reliably also during a single failure.

References: SÚJB BN-JB-1.0 Safety Guide (69, 81, 118)

In order to protect the pressure and cooling reactor circuit, requirements and conditions shall be defined for the operation and for tests of the system. This implies that the effects that may damage the circuit shall be analysed, including the highest permissible levels of static and dynamic pressures, temperatures, hydraulic or mechanical loads/stresses and pressure/temperature transients. Acceptance criteria shall be defined. The design solution of the circuit itself and of its auxiliary, control and protection systems shall ensure that the criteria are met with an adequate margin in any state accounted for in the design.

The safety system removing the residual heat produced from disintegration of fission products and accumulated heat from components that also operates independently on the nuclear installation's external power sources shall provide heat removal after shutdown and during the following time so that no design fuel limits and limits for the reactor's pressure and cooling units are violated, even under the conditions of simple failure and, at the same time, if one of the system's parts is not serviceable because of maintenance. The system shall efficiently prevent personnel exposure to ionising radiation and penetration of radioactivity into the environment. To this end, the system shall be equipped with appropriate means, including means for monitoring those functions.

In order to ensure the function of the safety systems for residual heat removal from the reactor core and the containment and for heat removal from nuclear safety-related systems during normal and abnormal operation, as well as in accident situations, the nuclear installation shall be equipped with an independent heat removal system reaching as far as the Ultimate Heat Sink, provided with an adequate degree of redundancy of the important equipments,as well as of power supply (see Section 3.8, Electrical systems), equipped with a system for detecting any penetration of radioactivity into the system and with means for preventing radioactivity leak into the environment.

References: IAEA SSR 2/1 Req. 70, 6.46, Req. 53

Supporting systems of the heat removal systems shall provide sufficient removal of heat from the nuclear power plant's components and systems, for which functionality is required in various operating regimes and under accident conditions.

The design of the heat removing systems shall support the option to separate the system's most critical parts.

The systems must enable transfer of residual heat from the nuclear power plant's safety-critical systems to the Ultimate Heat Sink. This function must be ensured with a very high level of reliability for all of the power plant's states.

References: WENRA App. E 7.3, 7.4

Criteria for protection of the pressure limit of the (primary) coolant shall be specified, including the coolant's maximum pressure, maximum temperature, and heat and pressure exertion.

If feasible, criteria for protection of the pressure limit shall also be applied for protection of the secondary cooling system.

3.9.1.2.1.1 Safety criteria

References: Decree No. 195/1999 Coll., Article 25 (2)

The nuclear safety-critical water systems, such as essential service water systems, shall be divided into several independent functional units. Such units shall be mutually functionally separated and, at the same time, connected with the option to disconnect the system's parts and detect leaks and capture them, so that the system works reliably even in case of a simple failure.

Systems and their nuclear safety-critical installation (must be) redundant in a necessary scale.

References: SÚJB BN-JB-1.0 Safety Guide (81)

The safety system removing the residual heat produced from disintegration of fission products and accumulated heat from components that also operates independently on the nuclear installation's external power sources shall provide heat removal after shutdown and during the following time so that no design fuel limits and limits for the reactor's pressure and cooling units are violated, even under the conditions of simple failure and, at the same time, if one of the system's parts is not serviceable because of maintenance. The system shall efficiently prevent personnel exposure to ionising radiation and penetration of radioactivity into the environment. To this end, the system shall be equipped with appropriate means, including means for monitoring those functions.

References: SÚJB BN-JB-1.0 Safety Guide (69)

In order to protect the pressure and cooling reactor circuit, requirements and conditions shall be defined for the operation and for tests of the system. This implies that the effects that may damage the circuit shall be analysed, including the highest permissible levels of static and dynamic pressures, temperatures, hydraulic or mechanical loads/stresses and pressure/temperature transients. Acceptance criteria shall be defined. The design solution of the circuit itself and of its auxiliary, control and protection systems shall ensure that the criteria are met with an adequate margin in any state accounted for in the design.

References: SÚJB BN-JB-1.0 Safety Guide (118)

In order to ensure the function of the safety systems for residual heat removal from the reactor core and the containment and for heat removal from nuclear safety-related systems during normal and abnormal operation, as well as in accident situations, the nuclear installation shall be equipped with an independent heat removal system reaching as far as the Ultimate Heat Sink, provided with an adequate degree of redundancy of the important equipments,as well as of power supply (see Section 3.8, Electrical systems), equipped with a system for detecting any penetration of radioactivity into the system and with means for preventing radioactivity leak into the environment.

Resistance to external impacts

Resistance of the nuclear safety-critical water systems to external impacts shall comply with respective requirement of Section "3.3.3 Protection from external impacts".

3.9.1.2.2 Water systems that are not nuclear-safety critical

The specified legislation does not contain the requirements for water systems that are not critical from the safety aspect. This shall be specified within the next stages of the licensing documentation.

3.9.1.3 OPERATION SUPPORTING SYSTEMS

This section describes basic legislative requirements for the design of safety-critical supporting systems.

Operation supporting systems consist of a set of selected systems that provide supporting functions for the primary circuit's main systems (e.g. supply of compressed air, chemical mode control, coolant drainage and replenishment, media sampling, coolant treatments, etc.)

References: IAEA SSR 2/1, Req. 69

The design of the supporting and auxiliary systems shall be solved so that execution of these systems is in accordance with safety importance of the supplied components or systems.

3.9.1.3.1 COMPRESSED AIR SYSTEMS

The compressed air systems serve the purpose of sources of compressed air with required parameters for the power plant's installation, such as pneumatic drives for emergency stop fittings that are located at the edge of the hermetic zone.

References: IAEA SSR 2/1, Req. 72

Design bases for the compressed air system that supplies nuclear safety-critical installation shall specify requirements for quality, flow capacity, and purity of supplied compressed air.

3.9.1.3.2 Process sampling systems and post-accident sampling systems

The function of the process and post-accident sampling systems is to provide information on process fluids from various plant systems in operational states and accident and post-accident conditions.

Such information is acquired from a radiochemical analysis of the collected sample.

References: SÚJB BN-JB-1.0 Safety Guide (83, 84)

(1) The process and post-accident sampling systems enable, as needed, to monitor, measure, and record operation parameters critical for provision of the nuclear safety, radiation protection, and accident management in normal and abnormal operation, as well as in accident conditions.

(2) In accident conditions, these systems enable, as needed, toprovide:

- Information on parameters and system states, which can affect the development of the fission reaction, damage to the fuel system, integrity of the primary circuit, the containment, and related systems so that accident management guidelines can be appliedBasic information on the development of the accident
- The information that enables the prognosis of the spread of radionuclides and ionising radiation to the vicinity of the plant.

References: IAEA SSR 2/1 Req. 71

The plant shall be equipped with process and post-accident sampling systems for timely determination of concentrations of selected radionuclides in fluid process systems and in gaseous and liquid samples collected from various systems and environments in all operation states and in accident conditions.

The plant shall have suitable means for monitoring of activity in process fluids that have the potential for significant contamination, and suitable means for collection of process samples.

The requirements for the radiation monitoring systems are specified in Sections "3.11.1.5 The systems of radiation monitoring of process fluids and effluents", "3.11.2.5 Systems of radiation monitoring of process fluids and effluents in the preliminary design concept", and "3.12 Radiation protection".

References: SÚJB BN-JB-1.0 Safety Guide (72, 110)

The process and post-accident sampling systems shall take into account requirements for isolation of individual plant systems.

The process and post-accident sampling systems shall enable (as needed) monitoring and evaluation of various monitoring programmes and plans:

- Monitoring of ionising radiation and radionuclides [Decree No. 195/1999 Coll. [L. 266], JB 1.0 Safety Guide [L. 276], Article 43, SÚJB BN-JB-1.0 Safety Guide (129, 130)]
- Monitoring of the equipment state and compliance with operational limits and conditions, [SÚJB BN-JB-1.0 Safety Guide [L. 276] (48)]
- Monitoring of integrity of barriers and detection of possible leaks, [SÚJB BN-JB-1.0 Safety Guide [L. 276] (70, 81, 118, 121, 126)]
- Monitoring of the chemical mode, [SÚJB BN-JB-1.0 Safety Guide [L. 276] (121)]

3.9.1.3.3 Drainage systems

The specified legislation does not contain the requirements for drainage systems. This shall be specified within the next stages of the licensing documentation.

3.9.1.3.4 Chemical and volume control system

The Chemical and volume control system ensures the optimum conditions for prevention of creation of corrosion products from construction materials of the primary circuit and coverage of fuel cells, including deposition of active corrosion products on the inner surface of the primary circuit.

The boric acid recovery system is a conceptually connected system.

References: Decree No. 195/1999 Coll., Article 24

(1) The coolant refilling system shall be capable to compensate leakage and fuel

volume changes under normal and abnormal operations, considering the coolant bypass for purification, in order that the stipulated design limits may be kept.

(2) The coolant purification system shall be capable to remove the corrosion products

and fission products which release from the fuel elements at their appropriate damages, and in the same time to keep the required purity parameters of the primary circuit.

References: IAEA SSR 2/1, Req. 50, 6.17

The capabilities of the necessary plant systems shall be based on the specified design limit on permissible leakage for the fuel, with a conservative margin to ensure that the plant can be operated with a level of circuit activity that is as low as reasonably practicable, and to ensure that the requirements are met for radioactive releases to be as low as reasonably achievable and below the authorized limits on discharges.

References: SÚJB BN-JB-1.0 Safety Guide (73, 80)

In order to maintain a sufficient amount of coolant in the reactor's pressure and cooling circuit and for regulation of coolant's volume changes during normal and abnormal operation, a coolant refilling system shall be designed.

The design shall be able to ensure coolant treatment to remove impurities and radioactive matters, including removal of corrosion and fission products, so that chemical mode criteria are met and operation with fuel leaks permitted by the design is enabled.

3.9.1.4 HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS

This section describes basic legislative requirements for the design of heating, ventilation, and air conditioning systems.

3.9.1.4.1 Basic requirements

References: Decree No. 195/1999 Coll., Article 44 (a); SÚJB BN-JB-1.0 Safety Guide (119); IAEA SSR 2/1 Req. 73, 6.48, 6.49

Ventilation and filtration system shall provide the following functions that will ensure suitable conditions for personnel, as well as installed equipment on premises where nuclear safety-critical installation is located, during all operation states, even under accident conditions, whereas such functions shall be specified in accordance with relevant regulations for workplace.

Basic functional system requirements are:

- Prevent unacceptable dispersion and/or uncontrolled leak of airborne radioactivity substances in the various nuclear installations, in accordance with the requirements for their accessibility
- Reduce concentration of airborne radioactive substances below the levels stipulated in the special legal decree (Decree No. 307/2002 Coll. [L. 4]), if radionuclides are dispersed and leaked into the nuclear installation's accessible areas
- Maintain the specified ambient conditions
- to keep the levels of airborne radioactive substances in the plant below authorized limitsto ventilate rooms containing inert gases or noxious gases without impairing the capability to control radioactive effluentsmaintaining an appropriate direction of air flow between different areas, such that flow is not directed from a volume with a higher potential for airborne radioactivity, to one of lower potential. Areas of higher contamination shall be maintained at a depression (partial vacuum) relative to areas of low contamination and other accessible areas.

References: Decree No. 195/1999 Coll., Article (b), IAEA SSR 2/1 Req. 79, 6.63

The ventilation and filtration system shall be equipped with reliable filters with sufficient capture efficiency, whereas their parameters and functions shall be continuously monitored for the duration of their service life. It shall be possible to replace filter fillings while simultaneously maintaining air flow through the system. It shall be possible to inspect efficiency of the filters.

References: Decree No. 195/1999 Coll., Article 44 (c)

Nuclear safety-critical installation shall be backed up so that the heating, ventilation, and air conditioning systems can reliably operate even under a simple installation failure.

3.9.1.4.2 Fire protection

References: Decree No. 195/1999 Coll., Article 9 (2)

Fire protection regarding the heating, ventilation, and air conditioning system shall be designed so that it corresponds with the overall concept of the power plant's fire prevention and is in accordance with requirements stipulated in Fire Prevention Programme (see Section 3.9.1.5.1).

References: WENRA Issue S 4.4

Heating, ventilation, and air conditioning systems shall be designed so that in case of fire, fire separation of individual segments is ensured.

References: WENRA Issue S 4.5

Components of the heating, ventilation, and air conditioning system (such as connecting pipes, air conditioning machine rooms, and filters) that are located outside of fire segments must show the same fire resistance as the respective segments or they must be able to ensure separation of the segment in question by suitably dimensioned fire dampers.

3.9.1.5 OTHER SUPPORTING SYSTEMS

This section describes basic legislative requirements for the design of selected supporting systems (in particular, lighting and auxiliary diesel generator systems). In accordance with RG 1.206 [L. 275], this Section also summarizes the basic legislative requirements for the fire prevention programme. In the Preliminary Safety Analysis Report that will follow, this Section will contain properties of design bases for safety-relevant components of the supporting systems, description of the systems, safety assessment, requirements for tests and instrumentation, control, blocks, and signalizations.

References: SÚJB BN-JB-1.0 Safety Guide (120)

The nuclear installation design shall define requirements for other auxiliary and supporting systems that provide critical services or media to maintain serviceability of the installation important to nuclear safety, such as power supply, environmental conditions, and operating media (water, compressed air, fuel, lubrication, gases, etc.).

3.9.1.5.1 Fire Prevention Programme

Implementing the "defence in depth" principle, the Fire Prevention Programme ensures sufficient fire prevention regarding nuclear safety-critical installation in all operating states of the nuclear installation, and also protection of staff, residents, and the environment from excessive radiation threat. Defined objectives for individual protection levels are as follows: Prevention of fire occurrence, otherwise fast detection, reporting, and extinguishing of fire that occurred despite preventive measures taken, and last but not least, prevention of the spread of not-extinguished fire with the objective of eliminating or reducing its negative impacts on the safety-function systems and preventing radioactive leak to the environment.

3.9.1.5.1.1 Basic principles of designing the fire protection system

References: Decree No. 195/1999 Coll., Article 9 (1), SÚJB BN-JB-1.0 Safety Guide (137)

The nuclear installation and option of operators' interventions must not be jeopardized by unreasonable risk of fire, explosion, or burning product. Therefore, also for this risk, the nuclear installation design shall apply safety principles and requirements stipulated in the special guide (SÚJB BN-JB-1.0 Safety Guide [L. 276]) and respect requirements of the general legal regulation (Act No. 133/1985 Coll., on Fire Prevention, as amended) and its implementing decrees, with special regard to the necessity to prevent or reduce spread of radioactive matters.

References: Decree No. 195/1999 Coll., Article 9 (1); SÚJB BN-JB-1.0 Safety Guide (138); WENRA Issue S 2.1, 2.2

Nuclear installations, especially nuclear safety-critical nuclear installations, shall be designed and sited so that fire occurrence probability in the locality is small, and installations are able to resist its affects so that they retain their ability to fulfill safety functions, especially the option to safely shutdown the reactor and maintain it in the subcritical state, conduct its residual output away for a sufficiently long time, and ensure reduction of radionuclide leaks and, at the same time, so that monitoring of the nuclear installation's state during and after the fire is ensured.

References: Decree No. 195/1999 Coll., Article 9 (2), IAEA SSR 2/1 Req. 74, 6.54

For nuclear installations, especially nuclear safety-critical installations, only non-flammable or reduced-flammability materials shall be used.

References: WENRA Issue S 2.3

Buildings containing nuclear safety-critical installations shall be divided into fire segments. Premises on which such installations will be sited shall show as low a fire burden as possible. Separation into fire segments shall be carried out so that redundant safety systems are mutually separated. Where division into fire segments is not possible, division into fire units shall be done based on a fire risk analysis, which will ensure fulfillment of required safety functions during a fire using active as well passive fire prevention means.

References: WENRA Issue S 2.4

Buildings containing radioactive materials that could leak into the environment in case of fire shall be designed so that such leaks are minimized.

References: WENRA Issue S 5.1

Within preventive measures, the nuclear installation's operator shall define procedures for minimization and control of amounts of flammable materials. At the same time, potential fire initiation sources that could affect nuclear safety-critical installations shall be minimized. For the purpose of ensuring serviceability of the fire prevention system, activities and procedures that will ensure inspections, maintenance, and tests of fire-separation structures and barriers preventing fire spread and of fire detecting and reporting systems, as well as fire extinguishers, shall be stipulated.

3.9.1.5.1.2 Protection against external and internal impacts

Protection of the nuclear installation against internal and external impacts and fulfillment of safety functions shall be solved in accordance with Sections 3.3.3 and 3.3.5 of this ISAR.

3.9.1.5.1.3 Fire risk analysis

References: Decree No. 195/1999 Coll., Article 9 (4), WENRA Issue S 3.1

For facilities that are important from the aspect of the nuclear safety-critical installations, fire risk evaluation shall be produced or a fire risk analysis shall be produced and continuously updated, and it shall prove that fire safety objectives have been met, suitable fire protection measures were designed, and all other necessary administrative measures were identified.

References: SÚJB BN-JB-1.0 2011 Safety Guide, (139), WENRA Issue S 3.2, 3.3, 3.4

The deterministic fire risk analysis shall be produced and supplemented with the probability analysis (1st level PSA), which shall also include:

- Normal operation situations and individual fires with subsequent spread anywhere where flammable material is present
- Evaluation of reliable combinations of fires with postulated initiation events that occurred independently of them
- Assessment of fires, as well as impacts of fire extinguishing system actions
- Assessment of sufficiency of fire-prevention, technical, as well as organizational measures

Requirements for such analyses are stipulated by the State Office for Nuclear Safety.

3.9.1.5.1.4 Fire prevention system

References: Decree No. 195/1999 Coll., Article 9 (3), SÚJB BN-JB-1.0 2011 Safety Guide (140), (141), IAEA SSR 2/1 Req. 74, 6.51, 6.52, WENRA Issue S 1.1, 4.1, 4.2

Nuclear installation facilities shall be equipped with:

- Systems detecting and signalising fire occurrence inform operators in the main control room of the fire prevention system's control centre equipped with backed up power supply and conducting a signal via cables that are functional during a fire
- Fixed and mobile fire extinguishing system designed so that even in case of its failure or random initiation, functional ability of equipment important for nuclear safety-critical installations

Non-flammable or not-easily flammable materials or a cabling solution that does not spread fire shall be used everywhere where it is justified, especially for nuclear safety-critical installations and also on premises of the operation and back-up control rooms.

References: Decree No. 195/1999 Coll., Article 9 (5)

Nuclear installation containing a nuclear reactor with its output exceeding 50 MWt shall be secured by the permit bearer's fire brigade from the building phase onward.

References: WENRA Issue S 4.3

Supply of water to fire hydrants outside, as well as inside of buildings, shall be designed based on the fire risk analysis, and it shall ensure suitable coverage of needs of all power plant nuclear safety-critical premises.

References: WENRA Issue S 4.4, 4.5

The ventilation system shall be designed and configured in accordance with Section "3.9.1.4 Heating, Ventilation, and Air Conditioning Systems" of this ISAR.

References: IAEA SSR 2/1 Req. 74, 6.50, 6.53

Based on the fire risk analysis, fire prevention systems shall be installed on premises of the whole power plant, including the fire detection and reporting system, fire extinguishing systems, fire-prevention barriers, fire prevention dividers (structures), and smoke exhaust systems.

Fire detection and extinguishing systems necessary for protection in case of fire caused by a postulated initiation event shall have resistance corresponding with effects of the postulated initiation event.

References: IAEA SSR 2/1 Req. 36, 5.64, 5.65, WENRA Issue S 2.5

The nuclear power plant shall be equipped with sufficient amounts of safe escape routes with distinct and permanent indication that shall be fixed with reliable emergency lighting, ventilation, and other equipment important for their safe use.

Escape routes shall comply with respective requirements of the Czech, as well as international legislations related to radiation and fire-prevention, as well as respective requirements of the Czech legislation related to occupational health and safety and physical protection.

The power plant design shall be solved so that any premises affected by internal or external impacts or other combination of events considered while designing the power plant shall always have at least one escape route available.

References: IAEA SSR 2/1 Req. 65, 6.39

The power plant design shall take such measures, including building of information channels and barriers between the control rooms and surrounding environment, which would ensure protection of operators in control rooms from risks related to fire and the presence of toxic explosive gases.

3.9.1.5.1.5 Organization of fire intervention

References: WENRA Issue S 6.1, 6.2, 6.4

The power plant's operator shall implement relevant measures for fire safety provision and control, in accordance with the fire risk analysis.

The operator shall produce and continuously update emergency plans, which shall clearly define activities and responsibilities of individual employees in case of fire in the power plant. A fire intervention plan shall be produced and continuously updated, and it will ensure coverage of all power plant premises where fire could occur and jeopardize nuclear safety-critical installations and also ensure sufficient protection of radioactive materials.

Requirements regarding organization, equipping, physical abilities and training of the power plant employees participating in fire intervention shall be documented by the nuclear installation's operator, and their sufficiency shall always be assessed in accordance with the national legislation.

References: WENRA Issue D 3.2

All operating staff of the power plant, including employees of contractor organizations, shall be informed about the basic principles of nuclear safety, radiation safety, fire safety, emergency planning, and occupational health and safety.

References: WENRA Issue R 3.3

The operator shall implement measures to provide technical support to the operating staff. Teams taking care of alleviation of results of emergency incidents related to radiation protection, inspection of damage to technical equipment, fire prevention, etc. shall be organized.

3.9.1.5.2 Communication systems

This issue is addressed within Section "3.3.1.1.17 Communication means".

3.9.1.5.3 Lighting systems

Lighting systems include operation (regular), back-up, and emergency lighting systems, including lighting of escape routes. The purpose of the lighting systems rests in ensuring required qualitative and quantitative lighting parameters on specific power plant premises for the periods of normal operation, abnormal operation, as well as under accident conditions in the power plant, in accordance with the top requirement to ensure nuclear safety and industry-specific Czech legislation requirements, including relevant hygienic and fire regulations.

References: IAEA SSR 2/1 Req. 75

The power plant shall be equipped with relevant operation and emergency lighting systems that will support safe operation on all the power plant's operating premises in the course of normal and abnormal operation, as well as under accident conditions.

References: Decree No. 195/1999 Coll., Article 31 (1)

The emergency lighting systems classified as class I accessories of backed up power supply must be fed continuously, whereas their supply must be provided from sources that supply energy immediately (a battery or a battery with current inverters).

References: IAEA SSR 2/1 Req. 36

The power plant shall be equipped with a sufficient number of safe escape routes with reliable emergency lighting. Description of the emergency lighting systems must be included in the fire risk analysis.

3.9.1.5.4 Supporting systems for diesel generators

Diesel generators can be used in the ETE3,4 design as emergency sources for backed up category II power supply networks (for more detailed description, see Sections 3.8.1.1 through 3.8.1.3) or as alternative alternating power supply sources for handling events of the Station Blackout type (for more detailed description, see Section 3.8.1.4).

Correct functioning of supporting technical systems is the condition for proper functioning of diesel generators, in accordance with specified safety requirements. Important supporting technical systems include the fuel storage and supply system, water cooling, and initiation system. The following legislation requirements are stipulated for the supporting systems:

References: IAEA SSR 2/1 Req. 68, 6.44, Req. 69

Construction and reliability of operation of diesel generators used as emergency sources for category II backed up supply networks, including respective supporting technical systems shall correspond with requirements of fed components, systems, and installations and their importance from the nuclear safety aspect.

References: IAEA SSR 2/1 Req. 68, 6.45

Design of diesel generators used as emergency sources for category II backed up power supply grids shall take into account requirements regarding provision of proper functioning of supporting systems, especially as follows:

- Ensuring operation of a diesel generator at a required output for the specified period of time (i.e. requirements for the fuel storage and supply system)
- Proper functioning of the water cooling system
- Ensuring the ability to execute initiation and operation under all required conditions and for the required period of time (i.e. requirements for the initiation system)

3.9.1.5.5 Lifting equipment

The power plant shall be equipped with suitable lifting equipment for transport of systems, structures, and components important from the safety aspect or transport of other loads on premises of the power plant close to the important safety systems, structures, and components.

References: IAEA SSR 2/1 Req. 76, 6.55

Design of the lifting equipment for transport of systems, structures, and components important from the safety aspect or transport of other loads on premises of the power plant close to the safety-important systems, structures, and components shall include:

- Technical measures preventing lifting of excessive loads
- Sufficiently conservative technical solutions preventing unintentionally initiated load release that could unfavourably affect safety-important systems, structures, and components
- Layout of premises that would enable safe movement of the lifting equipment, as well as transport of a suspended load

Design of the lifting equipment for transport of systems, structures, and components important from the safety aspect or transport of other loads on premises of the power plant close to the safety-important systems, structures, and components shall be:

- Used in specific power plant modes only; otherwise, they will be properly secured
- Seismic-qualified in the corresponding manner

3.9.2 PROPERTIES OF THE SUPPORTING SYSTEM DESIGN FOR THE PURPOSE OF THE PRELIMINARY ASSESSMENT

This section describes the properties of the design of the supporting systems for the preliminary assessment of the design concept. The information for the specification of the design properties was based on the technical part of the tender documentation stipulating the requirements for safety and technical design of the future power plant design. This section includes partial assessment as to whether the tentative concept of the design segment in question complies with legislative requirements specified in Sections 3.9.1.1 through 3.9.1.5. The scope of this section includes identification and assessment of the general level of the requirements for the functions of the supporting systems, whereas particulars of the specific technological implementation of such systems shall be included in the design documentation of the selected supplier of the NPP, and the assessment will be performed within the next stage of the safety documentation.

Basic requirements for a design solution of the supporting systems is based on the general requirement for ensuring a sufficient level of support of the technical systems, for support of which the supporting systems are intended. The supporting system design solutions shall be designed so as to ensure satisfaction of the basic safety requirements and principles specified in Section 3.3.1. These include, in particular, the principle of defence in depth and the design criteria specified for the systems, structures, and components within the design requirements, including the relevant design requirements based on the conditions of the site, as specified in Chapter 2 and summarized for the most critical ones in Section 2.10.

Supporting systems that support safety-critical equipment shall be classified on the same level as the supported systems.

The design shall include the following selected supporting systems, whereas the particular method selected for implementation does not have to include all systems, structures, and components specified in this Section; however, if the selected design utilizes them, they shall fulfill applicable requirements specified in this Section.

The design shall respect the principle that if the performance of safety functions requires the functioning of a specific system, it also requires the functioning of its respective support systems. The functionality of such supporting systems has to be preserved during all the initiating events assumed by the design.

3.9.2.1 FUEL HANDLING AND STORAGE IN THE TENTATIVE DESIGN CONCEPT

This section describes basic summaries of the tentative design concept of the installation for handling of fresh, irradiated, and spent fuel and for its storage. For such systems, the concurring license documentation level structured in accordance with [L. 275] shall contain design bases for the most important components, description of the systems, safety assessment, assessment of reliability and scope of inspections and tests focused on a proof of provision of a sufficient sub-critical level for all normal and abnormal conditions that can occur within the systems.

3.9.2.1.1 Handling of fresh nuclear fuel and its storage in the tentative design concept

The fresh fuel handling and storage system shall be designed so that it fulfills the following functions:

- Maintaining fuel in the sub-critical state
- Protection of the fuel system against damage
- Maintaining an acceptable radiation dose in work areas
- Provision of a site and equipment for fuel assembly inspections
- Receipt, inspections, and storage of new fuel assemblies
- Filling the reactor with fuel

Fuel stored in a wet or dry storage area (fresh or spent fuel) shall be kept in the sub-critical state (Kef lower than 0.95 in clean water), even under hypothetical accident conditions.

For fresh fuel with the highest enrichment stored in a dry storage area and under the optimum moderation conditions, the Kef value shall not exceed 0.98.

In ETE3,4, fresh fuel assemblies shall be stored in a dry storage area with a sufficient capacity for fuel replacement during shutdown.

The fresh fuel handling and storage system shall be designed so that it provides a site and equipment for reconstructions, repairs, and inspections.

Lifting and handling means, including transport and lifting equipment, shall be designed so that they prevent fuel damage as a result of a fall, crash, simple failure, power supply loss, or earthquake. Design of the power supply and protection of lifting and handling equipment shall provide for sufficient diversification and segregation.

3.9.2.1.2 Handling of irradiated and spent nuclear fuel and its storage in the tentative design concept

The spent fuel handling and storage system shall be designed so that it fulfills the following functions:

- Maintaining fuel in the sub-critical state
- Protection of the fuel system against damage
- Maintenance of containment insulation (for a transport channel if a fuel storage pool is located outside of the containment)
- Maintaining an acceptable radiation dose in work areas
- Provision of a site and equipment for fuel assembly reconstructions, repairs, and inspections
- Heavy component lifting
- Spent fuel storage, inspections, and transport preparation
- Filling and removing fuel from the reactor
- Storage of other irradiated internal reactor components before their transport outside of the unit
- Enabling of identification of seeping fuel assemblies and, after a repair, enabling a test of whether the repair was successful

Fuel stored in a wet storage area shall be kept in the sub-critical state (Kef lower than 0.95 in clean water), even under assumed accident conditions.

Residual heat released by stored spent fuel in pools shall be removed during all operating states, as well as under accident conditions.

Radiation protection shall be ensured by a sufficient water height above fuel assemblies

The radiation protection issue is described in Section "3.12 Radiation protection".

The fresh fuel handling and storage system shall be designed so that it provides a site and equipment for reconstructions, repairs, and inspections.

Lifting and handling means, including transport and lifting equipment, shall be designed so that they prevent fuel damage as a result of a fall, crash, simple failure, power supply loss, or earthquake. Design of the power supply and protection of lifting and handling equipment shall provide for diversification and segregation.

Heavy loads shall not be handled above the spent fuel pool.

Seeping fuel assemblies shall be stored in specially constructed hermetical casings. Seeping fuel assemblies shall be stored in the pool for storage of spent fuel.

The pool shall provide a sufficient area available for repairs and inspections of fuel assemblies, including the necessary repair and inspection equipment.

The system shall be equipped with filling, drainage, and chemical mode control.

As a result of a failure of any connected system, the pools shall not be drained below the minimum allowed level. Piping systems shall be designed so that no siphon is created in case of damage to the piping or a component of any connected system.

Technical monitoring and pool management systems shall measure a water surface level, temperature, and activity, and concentration of homogenous neutron absorbates. Control and display of measured data shall be possible at the main control room during operating states, as well as under accident conditions.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for fuel handling and storage, as described in this Section "3.9.2.1 Fuel handling and storing in the tentative design design" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 46, 47, of document IAEA SSR 2/1 [L. 252] Req. 80, 6.64, 6.65, 6.66, 6.67, 6.68 and SÚJB BN-JB-1.0 Safety Guide [L. 276] (121).

3.9.2.2 WATER SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

This section presents a basic summary of the tentative concept design and requirements of the technical implementation method of water systems.

In the Preliminary Safety Analysis Report that will follow, this Section will contain properties of design bases for safety-relevant components, a description of the system, safety assessment, requirements for inspections, tests, and instrumentation, control, blocks, and signalizations.

3.9.2.2.1 Nuclear safety-critical water systems in the tentative design concept

Essential Service Water System

From the aspect of the design concept, the essential service water system shall consist of:

- Pumping station with pumps providing circulation of coolant for heat removal
- External pipe distribution, including the Ultimate Heat Sink
- Internal distribution to appliances with Isolation valves for individual appliances
- Cooling water treatment facility (chemical dosing or water treatment using filtration)
- Additional water replenishment system and the blowdown or drainage system
- Technology control and management system with the option to execute checks and tests of selected equipment during operation

The essential service water system shall consist of mutually independent circulation circuits, i.e. independent divisions (system sections), of which each shall consist of the aforementioned units. The system shall be designed for cooling of heat sources of safety systems and nuclear safety and radiation protection-critical systems especially in the reactor reactor building in normal and abnormal operation, as well as under accident conditions. The essential service water system can also utilize some other installations of the power plant for cooling.

The cooling medium shall be fed by pumps from individual Pumping station to heat exchangers and coolers of various connected technical systems, including cooling of bearings of important engines and pumps that are essential for the unit's function. Pumps in the Pumping station shall be sufficiently backed-up, including their power supply. Selected systems shall be equipped with detection of penetration of radioactive matters and means for stopping their penetration to the environment. Warmed water shall then be supplied to heat absorbers and then returned to the Pumping station. With regard to evaporation and other planned losses, water replenishment shall be provided.

The essential service water system shall be designed so that any possible damage by raw water is prevented. The system shall contain equipment for easy inspection and maintenance of piping, technical equipment and its draining, and it shall also enable cleaning and replacement of selected components during operation. It shall be possible to use unused system parts for filling and drainage of treated water.

3.9.2.2.1.1 Safety criteria in the tentative design concept

Requirements for the essential service water system:

- The essential service water equipments shall be equipped with an independent system for heat removal up to the Ultimate Heat Sink. The quantity of independent systems and their capacity shall be designed so that the system will be able to provide removal of residual heat under all operation and accident conditions
- Under the maximum heat load, the system capacity shall be sufficient to safely maintain the nuclear installation in the shutdown state for at least 30 days even under the most conservative climatic conditions
- The essential service water Pumping station shall be constructed so that they enable independent functioning of each of the partial subsystems. Technology installed into all subsystems shall be identical and backed-up. Such redundancy principle shall be applied to the whole essential service water system, especially for cooling of appliances of nuclear safety-critical installations
- External pipe distributions to the production unit shall be designed so that there is no interconnection among individual independent system divisions, unless improvement of the system's safety, serviceability, or availability is proved
- Every essential service water system division shall be able to monitor and by means of regulation also effect the volume of heat removed from the connected systems and monitor and signal states of important installations and the system's operation parameters

- It shall be possible to monitor and control chemical mode and activity of the cooling medium in every division. The system shall be equipped with detection of penetration of radioactive matters and means for stopping their penetration to the environment.
- The essential service water system and its individual divisions shall be equipped with equipment for tests of operability of important installations, equipment for quick replacement of some piping components, and also equipment for flushing and cleaning the system
- The essential service water system shall be operated using such operation parameters to prevent possible penetration from connected technical systems, especially from the component cooling water system in the reactor building

3.9.2.2.2 Water systems not critical for nuclear safety in the tentative design concept

The specified legislation does not contain the requirements for water systems that are not critical from the safety aspect. This shall be specified within the next stages of the licensing documentation.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for the water systems, as described in this Section "3.9.2.2 Water systems in the tentative design concept" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 8 (1), Article 25 (2), of document IAEA SSR 2/1 [L. 252] Req. 70, 6.46, Req. 53, and SÚJB BN-JB-1.0 Safety Guide [L. 276] (69), (81), (118) and WENRA [L. 27] App. E 7.3 and 7.4.

3.9.2.3 OPERATION SUPPORTING SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

This section contains the basic summary of the tentative design concept and requirements for design solutions of selected supporting safety-critical systems that provide supporting functions for the primary circuit's main systems (e.g. supply of compressed air, chemical mode control, coolant drainage and , media sampling, coolant treatments, etc.)

In the Preliminary Safety Analysis Report that will follow, this Section will contain properties of design bases for safety-relevant components, description of the system, safety assessment, requirements for inspections, tests, and instrumentation, control, blocks, and signalizations.

3.9.2.3.1 COMPRESSED AIR SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

Autonomy of the compressed air systems serving safety systems shall be provided in the following scope:

- At least 6 hours from the accident beginning for the purpose of prevention of damage to the active zone in the case of design extension conditions
- At least 24 hours from the accident beginning for the purpose of retention of the containment's functionality in the case of design extension conditions

- At least 72 hours from the accident beginning in the case of basic design accidents

The compressed air systems shall be designed with a sufficient margin to ensure a possibility to carry out a sufficient scope of work during shutdown.

Diesel generators of safety systems shall be equipped with a back-up air initiation system that will enable at least 5 successful initiations without depletion of the pressure tank.

Pneumatic drives shall be equipped with pressure tanks to increase reliability of drives that shall be designed for a sufficient number of work cycles. Pressure tanks shall be equipped with a system that will ensure pressure retention in case of power supply outage in the compressed air supply system (usually a backflow valve).

3.9.2.3.2 Process sampling systems and post-accident sampling systems in the preliminary design concept

Process and post-accident sampling systems shall perform the following functions:

- Collect, condition (decrease pressure and temperature as needed), and transport representative samples to one or more sampling stations/boxes
- Monitor chemical and radiochemical parameters of process fluids
- Provide additional information on activity of reactor coolant
- Monitor boron concentration in reactor coolant
- Provide safe collection and handling of radioactive samples
- If needed, serve as a barrier against leaks of radionuclides

Sampling lines of the process and post-accident sampling systems connected to the systems located inside of the primary containment shall be equipped with automatic isolation features.

3.9.2.3.3 DRAINAGE SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

In technologically justified cases, system pipings shall be configured so that sufficient room for drainage and venting is provided. Drainage and venting routes shall end in such a way that no staining of installations or walkways occurs.

Measures for complete capturing of leaks shall be implemented in points where a risk of leak of radioactive media is possible.

If big tanks containing radioactive media are closed in waterproof cells or walled in, such measure would be sufficient to prevent leaks.

3.9.2.3.4 Chemical and volume control system in the tentative design concept

Coolant volume control in the primary circuit shall be possible to implement under all normal operation conditions, especially during output changes and the reactor initiation and shutdown.

Monitoring of the primary coolant's chemical composition shall be provided by means of continuous measuring and sample collection.

The primary coolant's chemical composition shall be controlled under all normal operation conditions (by chemical dosing and degassing).

During all normal operation states and especially during fuel transfer, concentration of fission and corrosion products in the primary coolant shall be controlled using demineralization and mechanical filtration of the primary coolant.

Chemical and volume control system shall consist, as the minimum, of the following subsystems:

- Replenishment pipe path or paths with a coolant supply tank and chemicals replenishment tank
- Replenishment pumps with incorporated fixtures and piping
- Drainage path or paths with after-cooling and coolant pressure decreasing and with equipment for coolant flow control
- Cleaning path or paths with a mechanical filter and demineralization equipment, feed of additional water with incorporated pumps and tanks, and feed of boric acid with incorporated pumps and tanks

Under normal operation, the chemical and volume control system shall perform the following functions:

- Required coolant volume control and retention in the primary circuit under all normal operation conditions, especially during output changes and the reactor initiation and shutdown.
- Control of slow reactivity changes by a change of concentration of boric acid in the coolant (e.g. because of fuel burning or production of fission products) under all normal operation conditions, especially during the reactor shutdown
- Control of the primary coolant's chemical mode (e.g. concentration of dissolved gases and pH)
- Ensuring the required chemical composition of the primary circuit suitable for venting commencement and opening of the primary circuit during after-cooling until the cold shutdown state
- Control of concentration of fission and corrosion products in the primary circuit shall be controlled using demineralization and mechanical filtration under all normal operation conditions, especially during fuel transfer.
- Provision of continuous replenishment of coolant and its collection from sealing of main circulation pumps (if pumps with shaft sealing are used)
- Pressure control in the primary circuit by draining in case of the primary circuit's complete filling with coolant
- Provision of supply of chemicals
- Provision of removal of precious gases from the primary circuit
- Provision of the storage capacity for boric acid solution
- Provision of replenishment of the primary coolant into supporting systems
- Provision of hydrogen dosing in the primary circuit

Under abnormal operation, the Chemical and volume control system shall perform the following functions:

- Provision of coolant supply into supporting sprinklers in the volume compensator if the main circulation pumps are out of operation
- Provision of coolant replenishment in the primary circuit in case of accidents with a very small loss of the primary coolant for prevention of initiation of the emergency cooling system
- Provision of the automatic boric acid replenishment to achieve the sub-critical state (e.g. in case of rupture of small pipes in the secondary circuit)

The Chemical and volume control system shall not perform the safety functions during basic design accidents.

The Chemical and volume control system shall be:

- Able to maintain the reactor in the sub-critical state under the cold shutdown conditions, even under the coolant temperature conditions corresponding with temperature of the surrounding environment, even under the conditions when the coolant temperature corresponds with the temperature of the surrounding environment, and considering a complete extension of the regulation organ with the highest weight
- Able to ensure the reactor's switching into the unit's cold shutdown state conditions after the reactor's shutdown
- Able to provide protection against unwanted decrease of concentration on boric acid in the primary coolant

The Chemical and volume control system shall be:

- Maintain the coolant amount in the primary circuit in the acceptable operating scope of the volume compensator under all normal operation conditions
- Ensure replenishment for compensation of temperature-caused volume change (shrinking) of the coolant during coolant cooling and replenishment in the primary circuit for compensation of operating leaks during normal operation
- Ensure draining for compensation of temperature-caused volume change (expanding) of the coolant during heating of the primary circuit and during output changes

The Chemical and volume control system shall be able to provide control of the primary coolant's chemical mode so that it reduces the amount of contents of corrosion and fission products to the ALARA level. At the same time, it shall ensure reduction of possibilities of presence of borate deposits in the active zone, and in that way reduce the possibility of occurrence of changes in output distribution in the active zone caused by this phenomena and possible local corrosion.

The high-pressure component of the Chemical and volume control system shall be sited inside the primary containment. In the case of exceptions to this rule, a proof of insignificance of the risk of possible bypass of the primary containment shall be produced.

Design of the coolant replenishment, treatment and recovery system shall minimize the possibility and alleviate results of gas entries into replenishment pumps' inlets.

The coolant replenishment, treatment and recovery system shall be redundant in order to ensure sufficient reliability.

The demineralization unit and filters shall be separable and shielded to enable access to the unit without a need to drain other parts of the system.

The Chemical and volume control system shall use at least two pumps, each of which shall be designed for the full flow of the coolant replenishment, treatment and recovery system.

In case of failure of sealing of the replenishment pumps, a radioactive leak (if any) shall be minimized.

The maximum concentration of stored boric acid shall be limited to 7,000 ppm (it corresponds with approximately 40 g/kg).

The boric acid storage system shall ensure sufficient temperature to prevent precipitation of borates.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for the operation supporting systems, as described in this Section "3.9.2.3 Operation supporting systems in the tentative design concept" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 24, Article 43, of document IAEA SSR 2/1 [L. 252] Req. 50, Req. 69, Req. 71, Req. 72, Req. 81, 6.17 and SÚJB BN-JB-1.0 Safety Guide [L. 276] (48), (70), (72), (73), (78), (80), (81), (83), (84), (110), (118),(121), (126), (129), (130).

3.9.2.4 HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS IN THE PRELIMINARY DESIGN CONCEPT

This section describes basic legislative requirements for the preliminary design concept of heating, ventilation, and air conditioning systems. In the Preliminary Safety Analysis Report that will follow, this Section will contain a description of the design basis for the safety-relevant components, and a description of the system, safety assessment, requirements for inspections, tests, and instrumentation, control, blocks, and signalizations.

The heating, ventilation, and air conditioning system in the power plant's areas and buildings shall be conceptually designed and equipped with appropriate equipment so that it fulfils specified requirements regarding safe operation of the power plant (in all operating states, even under accident conditions) in relation to the environment, as well as in relation to personnel. It concerns appropriate; designed heating, ventilation, and air conditioning and other HVAC systems. The basis for the conceptual design of the system will be the purpose of the object (area)from the aspect of operation of the installed technical equipment and method of its operation by employees.

In order to maintain a suitable environment (temperature, pressure, relative humidity, clean air), the designed heating, ventilation, and air conditioning systems shall fulfil the following functions for personnel and installed equipment:

- removal of heat generated by equipment
- renewing air
- maintaining an appropriate direction of air flow between different areas, such that flow is not directed from a volume with a higher potential for airborne radioactivity, to one of lower potential
- reducing the concentration of radioactive gases and aerosols, removing and inhibiting the spread of contamination by providing appropriate filtration systems
- maintaining appropriate relative pressure within a building or volume to assure controlled leakage of potentially radioactive effluents
- Ensure required internal air conditions for employees, as well as for technical equipment
- preventing migration of fire, smoke, hot gases, and fire suppressants into other fire areas

Internal air parameters shall be maintained in accordance with the requirement for provision of correct and reliable operation of equipment and compliance with required hygienic conditions necessary for personnel.

Fire prevention regarding the heating, ventilation, and air conditioning system shall be designed so that it corresponds with the overall concept of the power plant's fire prevention. More details regarding the fire prevention issue are shown in Section "3.9.1.5.1 Fire prevention programme" and "3.9.2.5.1 Fire prevention programme in tentative design concept".

Partial preliminary assessment

The preliminary concept of the design summarizing the key requirements for the air conditioning systems, as described in this Section "3.9.2.4 Heating, ventilation, and air conditioning systems in the tentative design concept" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 44 (a, b,c), of document IAEA SSR 2/1 [L. 252] Req. 73, 6.48, 6.49, 6.4, Req. 79, 6.63, Req. 74, WENRA [L. 27] Issue S 4.4, S 4.5 and SÚJB BN-JB-1.0 Safety Guide [L. 276] 119.

3.9.2.5 OTHER SUPPORTING SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

This section describes basic summary of the tentative design concept and requirements for the design of selected supporting systems (in particular, lighting and auxiliary diesel generator systems). In accordance with RG 1.206 [L. 275], this Section also summarizes requirements for the fire prevention programme in a form that enables formulation of the tentative design concept.

3.9.2.5.1 Fire prevention programme in the tentative design concept

This section contains properties of the fire prevention programme in the tentative design concept. The information for the specification of the design properties from the aspect of the fire prevention programme was based on the technical part of the tender documentation stipulating the requirements for safety and technical design of the future power plant design.

In relation to producing a fire risk analysis, the design concept shall ensure fulfillment of sufficient fire prevention regarding nuclear safety-critical installation in all operating states of the nuclear installation, and also protection of staff, residents, and the environment, in accordance with requirements of the Czech legislation. The fire risk analysis shall ensure collection of basic information on configuration of the power plant, its equipment and used materials, as well as division into fire segments and zones. Based on this information, necessary measures needed for fulfillment of safety functions in such fire segments shall be identified, and assessment of adequacy of fire prevention with regard to the nuclear safety and fulfillment of probability safety objectives shall be carried out. Fulfillment of such objectives in the nuclear power plant design shall be achieved by using passive, as well as active fire prevention components.

Within passive measures, the design concept shall make the maximum use of non-flammable, not easily flammable, or materials that do not spread fire. Where such materials cannot be used, the design shall specify measures to prevent fire occurrence and its subsequent spread. With a sufficient rate of technical details, the design shall solve construction of cabling and cable-supporting systems, their location, and fire separation using fire barriers with redundancy maintained.

The design concept specifically emphasizes siting of nuclear safety-critical installations or their siting in separate fire segments, separation by fire separation structures with required fire resistance, and especially minimization of fire occurrence probability in the given sections and its further spreading. Premises housing oil management or storage areas with larger amounts of flammable materials shall be solved in the design in a similar fashion. Even here the design concept shall consider decrease of fire load by separation into individual fire segments or zoning using fire barriers.

Within implementation of active fire prevention elements, the design concept shall consider utilization of the electrical fire signalization system for all premises of the nuclear plant. The fire signalization system, including automatic direct fire detection components, shall ensure initiation of fire alarm and cooperation with the alarm system providing information on a fire location to the relevant physical protection units and plant security guards. These systems shall be equipped with a backed-up power supply and the signal conducted by cables and cable paths with a functional integrity during fire. The design concept shall also solve placement and design of the fire-prevention system, such as fixed and half-fixed extinguishing equipment, automatic explosion-preventing equipment, and smoke and heat exhausting equipment. A fixed fire extinguishing system shall be designed so that even in the case of its failure or random initiation, functional ability of equipment important for nuclear safety-critical installations is not compromised.

The design shall solve escape routes, as well as access paths, ascending areas, and intervention paths used to carry out effective fire extinguishing intervention, all in accordance with the Czech legislation.

Partial preliminary assessment

The tentative design concept summarizing the key requirements for the fire protection programme, as described in this Section "3.9.2.5.1, Fire prevention programme in the tentative design concept" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 9 (1 - 5), SÚJB BN-JB-1.0 Safety Guide [L. 252] (137 - 141), of document SSR 2/1 [L. 276] Req. 36, 5.64, 5.65, Req. 65, 6.39, Req. 74, 6.50, 6.51, 6.52, 6.53, 6.54 and document WENRA [L. 27] Issue S 1.1, S 2.1-2.5, S 3.1-3.4, S 4.1-4.5, S 5.1, S 6.1, 6.2, 6.4; Issue D 3.2; Issue R 3.3.

3.9.2.5.2 COMMUNICATION SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

This issue is addressed within Section "3.3.1.1.17 Communication means".

3.9.2.5.3 LIGHTING SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

The lighting system design shall be produced so that the lighting systems provide required qualitative and quantitative lighting parameters on specific power plant premises for the periods of normal operation, abnormal operation, as well as under accident conditions in the power plant, in accordance with the top requirement to ensure nuclear safety and industry-specific Czech legislation requirements, including relevant hygienic and fire regulations.

The nuclear power plant design shall include work (regular) lighting, whereas selected facilities, premises, and technical installations shall be equipped with the backup lighting fed from category II backed-up power supply sources. The power plant design shall also include emergency lighting systems, including lighting of escape routes of specified construction facilities. Such systems shall be fed from category I backed up power supply systems with uninterrupted power supply from batteries or batteries with inverters.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for the lighting systems, as described in this Section "3.9.2.5.3 Lighting systems in the tentative design concept" creates preconditions for compliance with the requirements of Decree No. 195/1999 Coll. [L. 266] Article 31 (1) a document SSR 2/1 [L. 252] Req. 36, 75.

3.9.2.5.4 DIESEL GENERATOR SUPPORTING SYSTEMS IN THE TENTATIVE DESIGN CONCEPT

The supporting system of diesel generators for safety systems shall ensure proper functioning of supported technical diesel generator systems during normal operation, abnormal operation, and under accident conditions of the power plant, in accordance with the top requirement for provision of the nuclear safety.

All diesel generators for safety systems shall be equipped with independent supporting systems that are not mutually shared, especially a fuel system, air intake and emission exhaust system, initiating air system, lubricating oil system, cooling water system, and electrical systems.

Supporting systems necessary for correct functioning of diesel generators for safety systems shall be sufficiently dimensioned, sufficiently protected from impact of internal and external effects, and shall enable periodical inspections and testing during operation.

The amount of fuel sufficient for uninterrupted operation of all diesel generators for safety systems with the nominal output for at least 72 hours shall be stored on premises of the power plant. Fuel tanks shall enable operation of diesel generators for safety systems with the nominal output for at least eight hours.

Diesel generators of safety systems shall be equipped with a backed up air initiation system that will enable at least 5 successful initiations without depletion of the pressure tank.

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for the diesel generator supporting systems, as described in this Section "3.9.2.5.4 Diesel generator supporting systems in the tentative design concept" creates preconditions for compliance with the requirements of SSR 2/1 [L. 252] Req. 68, 6.44, 6.45, Req. 69.

3.9.2.5.5 Lifting equipment in the tentative design concept

The lifting equipment intended for transport of systems, structures, and components important from the safety aspect or transport of other loads on premises of the power plant close to the safety-important systems, structures, and components shall be designed, installed, operated, and tested in accordance with effective technical standards and corresponding Czech safety regulations for lifting equipment.

Such pieces of lifting equipment shall be designed so that:

- They lift and transport loads with reliability compatible with the risk related to such operation, especially with damage that can be incurred as a result of the load's falling
- They ensure as short handling times as possible, provided they remain compatible with all safety objectives
- They provide the minimum requirements for operators (remote control, a camera system for operators outside of the nuclear area)
- They have corresponding seismic qualification

Partial preliminary assessment

The tentative concept of the design summarizing the key requirements for the lifting systems, as described in this Section "3.9.2.5.5 Lifting systems in the tentative design concept" creates preconditions for compliance with the requirements of document SSR 2/1 [L. 252] Req. 76, 6.55.

3.9.3 COMPREHENSIVE PRELIMINARY ASSESSMENT OF THE SUPPORTING SYSTEMS

The ETE3,4 design shall meet the safety requirements specified in Section "3.3.1.1.1 Basic legislative requirements for provision of safety", as well as the requirements specified in Sections 3.9.1.1 through 3.9.1.5, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and related implementing decrees regarding the nuclear safety, radiation protection, and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270] documents.

To comply with the aforementioned requirements for systems, structures, and components and their equipment, the ETE3,4 design shall use specific technical systems equipped with corresponding supporting and auxiliary systems. Proper and reliable operation of the backup systems shall secure proper operation of the supported technological systems that shall sufficiently reliably support the specified technological and safety functions in terms of quality and quantity.

The principles of the design solution described in Section "3.9.2 Properties of the backup system design for the preliminary assessment purposes" were set up in line with the requirements that the license applicant imposed on potential suppliers of the nuclear installation during the tender, and they represent the design solution concept of this part of the design. The partial assessments performed proved that the expected design of selected supporting systems creates prerequisites for compliance with the relevant requirements for systems, structures, components, and safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], SÚJB BN-JB-1.0 Safety Guide [L. 276], IAEA SSR 2/1 [L. 252], and WENRA [L. 27].

The particular method of technical implementation of individual requirements for operation of the supporting systems specified in Section 3.9.2 shall be specified in detail only in the design documentation of the selected nuclear power plant supplier.

3.10 STEAM AND POWER CONVERSION SYSTEM

This comprehensive part of the Initial Safety Analysis Report is structured into three basic sections:

The introductory Section 3.10.1 summarizes, analyses, and specifies basic legislative requirements for steam and power conversion system while focusing on safety-relevant components of the secondary circuit.

The following Section 3.10.2 contains the description and specification of the basic requirements for the steam conversion system functions, which comprise an envelope of parameters and design requirements for this system and serve as basis for the determination of design characteristics for the needs of partial preliminary assessment.

The final Section 3.10.3 contains a summary preliminary assessement of the concept of the steam and power conversion system design, summarizing the conclusions of the partial preliminary assessments presented in Section 3.10.2. The assessment of the summarized requirements contains a preliminary design concept assessment required by the law.

At the next stage of the licensing documents the licence applicant shall submit evaluating and supporting information, allowing assessment of steam and power conversion system capability to meet its safety-relevant functions throughout the life time of the nuclear unit in all modes within the design conditions, including steady and operating transient modes, also during the occurrence of design extension conditions. The section shall also be supplemented with more detailed information at the depth and in the structure described in RG 1.206 [L. 275].

3.10.1 BASIC LEGISLATIVE REQUIREMENTS FOR STEAM AND POWER CONVERSION SYSTEM

3.10.1.1 GENERAL REQUIREMENTS FOR STEAM AND POWER CONVERSION SYSTEM

This section contains the basic national and international legislative requirements for steam and power conversion system design. In the following stage of the licensing documents, in the Preliminary Safety Analysis Report, the individual sections, except for the description of function of the secondary circuit and basic design characteristics, shall also contain the description of important design and operational values of the system and its major components.

Reference: Decree No. 195/1999 Coll.: Section 28(a), Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (123)

The system is designed to convert steam energy from the steam generators to steam turbine kinetic energy and subsequently to electrical power produced by the generator. The associated system function is to provide condensation of steam after its passage through the steam turbine and to remove heat to the cooling circuit. Through its parts, included in the appropriate safety categories, the system shall have the capability to provide a safe cooling down of to the unit during its planned and emergency shutdown, i.e. it shall reliably remove heat from the primary circuit in cooperation with other cooling systems listed in Sections "3.6.1.3 Emergency Core Cooling System" and "3.9.1.2 Water Systems".

Reference: Decree No. 195/1999 Coll.: Section 28(b), Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (123), (126), IAEA SSR 2/1 Req. 77

Secondary circuit design shall ensure that the specified design criteria shall be met in normal and abnormal operation as well as under accident conditions and that the relevant limits for the core and reactor pressure system shall not be exceeded. Any leaks from the primary circuit to the secondary circuit shall be detectable and if any such leak is detected, its further spreading shall be limited so as not to exceed the specified limits for exposure of operating personnel and for the population, and the specified limits for discharges of radionuclides to the environment.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (122)

A functional and physical interface shall be defined between the major equipment intended for energy conversion (steam and feed-water piping and turbine generators including I&C system) and the reactor part or the power grid to uniquely define the limits for graded requirements for equipment (classification and quantification), individual design principles and standards, reliability, quality, materials, etc.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (125)

The parts of the energy conversion system included in the selected equipment shall be designed, manufactured and tested in such a quality to meet the specified safety functions and to minimize the probability of leaks, fast-spreading defects or breaks to extremely low value. Such equipment shall be protected against internal as well as external influences. Therefore, criteria shall be specified to provide protection to the pressure part of the relevant systems, including maximum values of static and dynamic pressures, temperatures, hydraulic or mechanical loads and seismic load, and the design shall include the appropriate measures to comply therewith.

3.10.1.2 REQUIREMENTS FOR TURBINE GENERATOR

This section contains the basic legislative requirements for turbine generator design. In the following stage of the licensing documents, in the Preliminary Safety Analysis Report, the section shall contain the characteristics of design concept, description of turbine generator systems, characteristics of the approach to ensure integrity of turbine generator rotating parts including specification of the scope of preoperational and operational inspections.

Reference: IAEA SSR 2/1 Req. 77, 6.58

The turbine generator (turbine and generator set) shall be equipped with a complex of mechanical and electrical protections and controls (e.g. overspeed protection, protection against excessive vibrations), which shall ensure stable function, failure prevention and performance of functions within the principle of defence in depth. Measures shall be adopted to minimise the influence of missiles from the turbine generator on the systems important to safety.

3.10.1.3 REQUIREMENTS FOR ADMISSION STEAM SUPPLY STEAM

This section contains the basic legislative requirements for admission steam supply system design. In the following stage of the licensing documents, in the Preliminary Safety Analysis Report, the section shall contain the characteristics of design concept, description of admission steam system, evaluation of system robustness and reliability, characteristics of water systems, focusing on reduction of the corrosion erosion and requirements for materials including specification of the scope of preoperational and operational inspections.

Reference: IAEA SSR 2/1 Req. 77, 6.56, 6.57, Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (124)

The steam piping system shall contain properly dimensioned isolation valves that shall reliably function during all operating modes as well as during accident conditions, and shall have a sufficient capacity to be capable of preventing abnormal operation from developing into accident conditions.

3.10.1.4 REQUIREMENTS FOR OTHER EQUIPMENT OF STEAM AND POWER CONVERSION SYSTEM

This section contains the basic legislative requirements for feedwater system design. In the following stage of the licensing documents, in the Preliminary Safety Analysis Report, the section shall contain the characteristics of design concept, description of systems, safety assessment, description of instrumentation and measurement and specification of the scope of preoperational and operational inspections of the most important auxiliary systems of the secondary circuit.

Reference: IAEA SSR 2/1 Req. 77, 6.57, Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (124)

The feedwater system shall have a sufficient capacity to be capable of preventing abnormal operation from developing into accident conditions and to meet this function, the system shall be equipped with qualified isolation and protection elements.

3.10.2 DESIGN CHARACTERISTICS FOR THE NEEDS OF THE PRELIMINARY ASSESSMENT

This section describes the characteristics of the design for the needs of the preliminary assessment of the design concept. The information for the specification of the design characteristics was based on the technical part of the tender documents stipulating the requirements for safety and technical design of the future power plant design. This section includes partial assessment as to whether the preliminary concept of the design segment in question complies with the requirements specified in Section 3.10.1. The scope of this section includes the determination and assessment of the general level of requirements for the performance of the steam and power conversion system. The specific technological implementation of this system shall be specified in the design of the nuclear power plant and assessed in the next stage of the safety documentation.

The steam and power conversion system shall be designed so as to ensure meeting of the basic safety requirements and principles specified in Section 3.3.1. These include, in particular, the principle of defence in depth and the compliance with design criteria specified for the systems, structures and components specified within the design basis, including the relevant design requirements based on the conditions of the site as specified in Chapter 2 and summarised for the most important ones in Section 2.10.

The design selected for implementation need not include all systems, structures and components specified in this section. However, if the resulting design uses any of the systems, this system shall comply with the relevant requirements specified in this section.

3.10.2.1 STEAM AND POWER CONVERSION SYSTEM IN THE PRELIMINARY DESIGN CONCEPT

The project concept shall ensure the following design characteristics.

The heat energy of the steam generated in the steam generators shall be converted to mechanical power in the condensing steam turbine. The mechanical power shall be converted to electrical power in a three-phase turbo-alternator connected with the turbine via coupling.

The steam generated in the steam generators shall be supplied by the connecting piping to the blocks of admission valves and then to the high-pressure cylinder. After moisture separation and reheating, the steam from the high-pressure cylinder shall be supplied to the low-pressure cylinder. The steam from the low-pressure cylinders shall be admitted to the main condensers to be cooled using water from the cooling tower circuit. The condensation system shall convert the exhaust steam from the low-pressure cylinders to turbine condensate and then deliver the turbine condensate by means of the condensate pumps through the low-pressure regenerative heaters to the feedwater tank/deaerator. Feedwater shall be deaerated in the feedwater tank/deaerator and after reheating in the high-pressure regenerative heaters, delivered by means of feedwater pumps through the feedwater collector and feedwater flow regulating valves to the steam generators. The turbine generator shall be equipped with bypass system for direct delivery of the live steam to the main condensers.

The design of the power plant shall ensure that the operational states, including possible failures in the main turbine generator systems of the power plant, shall not result in exceeding of the design limits of fuel cladding and pressure boundary of the reactor coolant system. This shall mainly apply to the case of full loss of turbine load and turbine trip.

The design of the power plant, including the equipment of the secondary circuit, shall ensure that the conditions of normal operation, possible failures or the sequence of possible failures shall not cause exceeding of the design limits. It shall be demonstrated, through the systems performing safety functions, that the effects of the failure states shall not result in the reduction of the capability of reactor core cooling, in the loss of reactivity control and in an unacceptable release of radioactivity to the atmosphere.

The design of the power plant shall be capable of ensuring the safe cooling down of the unit during planned and emergency shutdown and heat removal from the reactor by means of natural circulation of reactor coolant and removal of steam to the main condenser, if the reactor shall be shut down.

The design shall ensure radiation monitoring in the secondary circuit. The radiation monitoring system shall ensure a continuous monitoring of retaining barriers and radioactive leaks. The radiation monitoring system shall ensure leak monitoring on the secondary side of steam generators. For more detailed information about the radiation monitoring system see Sections "3.11.1.5.4 Operating Media and Discharge

Radiation Monitoring Systems" and "3.11.2.5 Operating Media and Discharge Radiation Monitoring Systems in the Preliminary Design Concept".

The design shall include the steam generator blowdown system designed to reduce spread of radioactive substances leaked from the primary circuit to the secondary circuit. The steam generator blowdown system shall be designed to maintain the permissible chemical parameters of water on the secondary side of the steam generator during all operational states. The steam generator blowdown system shall be designed to take and treat water from any steam generator, if necessary. The blowdown system shall remove blowdown water from the place, where deposition of impurities is expected. In the event of radioactivity detection on the secondary side of the steam generator, the blowdown waste can be delivered to the liquid radioactive waste processing system.

Steam and Power Conversion System Materials

Only such materials shall be used in the manufacture of all equipment of the secondary circuit that have proved to be suitable for this purpose and comply with applicable codes, technical standards or technical conditions.

A theoretical calculation and an experimental verification shall demonstrate sufficient dimensioning of the individual components. The calculation shall also include a margin accounting for possible degradation of material properties that may take place during operation due to erosion, corrosion, fatigue, chemical environment and ageing, and a margin accounting for the uncertainty of the identification of the initial condition of the components and rate of degradation of their properties.

The selection of suitable material for each piece of equipment shall ensure and guarantee that no damage shall occur due to:

- High velocities of delivered media
- Corrosion and erosion damage
- Excessive distortion
- Plastic instability
- Elastic or elastic-plastic instability
- Gradual distortion
- Fatigue (formation and growth of cracks)
- Rapid fracture
- Thermal ageing

The design of the installation shall also take into account all the possible combinations of the various above mentioned loads (velocity, temperature, pressure, etc.).

The selected materials shall be suitable and resistant to erosion effects of the individual media, which they shall come into contact in all operating modes.

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for systems, structures and components, safety and technological functions of the steam and power conversion system specified in Section "3.10.2.1 General Requirements for

Steam and Power Conversion System in the Preliminary Design Concept" creates preconditions for meeting the requirements laid down in Decree No. 195/1999 Coll. [L. 266] Section 28(a, b), Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276] (122), (123), (125), (126) and IAEA document SSR 2/1 [L. 252] Req. 77.

3.10.2.2 TURBINE GENERATOR IN THE PRELIMINARY DESIGN CONCEPT

The turbine generator shall comprise of steam turbine, generator and the relevant auxiliary systems. After passing through the stop and control valves, the live steam shall be admitted to the high-pressure turbine cylinder. A moisture separator/reheater shall be included after the high-pressure cylinder from where the reheated steam free of moisture shall be parallelly delivered to several low-pressure cylinders. Some designs can additionally use an intermediate-pressure cylinder included before the low-pressure cylinders. After passing through the low-pressure cylinders, the steam is condensed in the condenser.

The turbine generator shall be equipped with the protection system providing protection and monitoring functions. The overspeed protection, lubricating oil pressure drop protection and condenser pressure increase protection shall be included among the important protection functions. The turbine shall be equipped with the fast power reduction system acting at design-defined exceeding of rated speed. The generator shall be equipped with the protection system, which shall include, but not be limited to, phase-to-phase fault protection, stator earth-fault protection, rotor earth-fault protection, asymmetric load protection, overvoltage protection, frequency protection, asynchronous operation protection, overcurrent protections and reverse current protections.The ranges defined by the design shall exclude the existence of the zones of critical speed of the turbine generator.

The possibility of turbine generator rupture and the occurrence of missiles shall be taken into account in designing the power plant layout and turbine generator orientation. The foundations of the turbine generator shall be designed with regard to the prevention of excessive vibrations in the zone of operating speed and maintaining the conforming machine alignment with an adequate rigidity under all operating conditions, including short circuits, as well as with regard to a sudden loss of electrical load and emergency shutdown. The bearings of the turbine generator shall ensure correct clearance between the rotors and the stator parts in all operational states. The turbine generator lubricating system shall ensure safe run-down of the set, even in the case of the loss of normal and standby power supply. The turbine shall be equipped with the sufficient number of appropriately located draining points to avoid damage to the machine caused by condensed steam.

The turbine bypass system shall remove the steam from the live steam piping directly to the condenser in those modes when the turbine shall not be capable of processing all steam generated in the steam generators. The turbine bypass system shall ensure:

- Maintaining the steam pressure in front of the turbine within the limits assumed by the design
- Turbine generator startup and shutdown
- Unit cooling down until the residual heat removal system can be put into the function, if the unit is equipped with this system

- Stable operation at low power level of the reactor (with connected or disconnected generator to or from the grid)

The opening time of the turbine bypass valves shall comply with the requirement for the prevention of technologically unjustified opening of steam generator safety valves. The closing time shall be sufficiently short to prevent excessive cooling of the primary circuit. The flow capacity of the turbine bypass valves shall be sufficient to prevent steam generator safety valves from opening after reactor trip from the full power and prevent pressurizer safety valves from opening after 100% loss of turbine load or turbine trip without simultaneous reactor trip.

The turbine bypass system shall enable the turbine to startup and the steam to be removed directly to the condenser after reduction in turbine generator load (e.g. when closing turbine stop valves), without tripping the reactor, while maintaining the level of steam pressure in front of the machine within the limits assumed by the design.

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for systems, structures and components, safety and technological functions of the steam and power conversion system specified in Section "3.10.2.2 Requirements for Turbine Generator in the Preliminary Design Concept" creates preconditions for meeting the requirements laid down in the IAEA document SSR 2/1 [L. 252] Req. 77, 6.58.

3.10.2.3 ADMISSION STEAM SUPPLY SYSTEM IN THE PRELIMINARY DESIGN CONCEPT

The live steam piping shall deliver the steam from the steam generators to the turbine stop valves and shall:

- Deliver the steam to the turbine in normal operation
- Deliver the steam to the turbine and to the turbine bypass piping during rapid load changes and during startup
- Deliver the steam to the turbine bypass piping in hot standby mode or during unit cooling down

The system shall isolate the steam generators from the point of steam line break by means of main steam isolation valves. The system shall provide protection of the steam generators against pressure increase. Drainage of all steam lines shall be provided. The relief valves that will be located between the containment and the main steam isolation valves shall be connected to the steam line. This relief system shall discharge the steam to the atmosphere and remove residual heat of the reactor when the turbine bypass system shall be out of operation. The equipment designed to remove steam to the atmosphere shall be qualified for water and steam discharge. If necessary, the equipment designed to remove steam to the atmosphere shall enable cooling down the primary circuit.

The concept of the admission steam supply system shall be designed to prevent abnormal operation from developing into accident conditions. The power plant shall have the capability of managing the turbine trip from 100% rated or lower power, without experiencing reactor trip or opening of steam generator safety valves. To achieve this capability, the short-term steam leakage through the relief valves to the atmosphere is permitted.

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for systems, structures and components, safety and technological functions of the steam and power conversion system specified in Section "3.10.2.3 Requirements for Admission Steam Supply System in the Preliminary Design Concept" creates preconditions for meeting the requirements laid down in the IAEA document SSR 2/1 [L. 252] Req. 77, 6.56, 6.57 and Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276] (124).

3.10.2.4 OTHER EQUIPMENT OF THE STEAM AND POWER CONVERSION SYSTEM IN THE PRELIMINARY DESIGN CONCEPT

The condensate system shall deliver the condensate from the condensers through low-pressure regenerative heaters to the feedwater tank/deaerator. The feedwater system shall deliver the deaerated feed water from the feedwater tank/deaerator through the high-pressure regenerative heaters to the steam generators. The parameters of the condensate and feed water shall be gradually increased when passing through the condensate and feed-water system. The unit shall be alternatively equipped with the emergency feed-water system (EFWS) providing feed-water for steam generators, unless the supply of operating feed-water is available, or with the system providing equivalent safety function by different design solution.

The individual feed-water lines to the steam generators shall be equipped with the feed-water isolation valves to mitigate the effects of pipe break. At least two identical pumps connected in parallel shall provide feed-water flow at full load. Each pump shall be equipped with an independent piping providing minimum flow. In addition to feedwater pumps, the design shall include at least one startup pump dimensioned to enable unit cooling down. The feedwater tank/deaerator shall be designed to make the level of removal of non-condensable gases compliant with the qualitative requirements for feed-water in all operating modes. The feedwater tank/deaerator and the suction part of the feedwater pumps shall be arranged to eliminate water boiling and cavitation in the pump suction line even in transient conditions.

The water supply in the feedwater tank/deaerator shall be sufficient to operate the feedwater pumps in normal operation and in transient modes. The normal water supply shall be sufficient to supply the feedwater pumps after shutdown of the condensate pumps, for sufficient period of time determined by the design. The initial water supply for standby supply, if the unit is equipped with this system, shall be sufficient to provide supply for at least 24 hours. The available water supply that shall be available in the power plant shall be sufficient to supply the steam generator or at least 72 hours.

The concept of the steam generator feedwater supply system shall be designed to prevent abnormal operation from developing into accident conditions. The shutdown of one feedwater or condensate pump shall not cause the shutdown of the steam generator or the reactor.

The feedwater system shall be designed to be able to automatically deliver the water with the required flow rate and parameters enabling to control the level in the steam generators during all operating modes and transients allowed by the design and to prevent them from developing into accident conditions. The feedwater system shall be capable of providing the feedwater supply to the steam generators when steam pressure increases due to the decrease of unit load from rated power. If the unit design assumes the use of the standby feedwater pump, it shall be dimensioned to enable to increase the power of the unit back to rated level after successful start of the standby pump. All steam generators at any power level shall be provided with the supply of feed water with the same temperature.

The design shall ensure that no bulk boiling or exposing of the reactor core occurs under the conditions of the design basis accidents (e.g. feedwater piping rupture) assuming that the main steam lines are automatically disconnected and the affected steam generator is isolated after 30 minutes.

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for systems, structures and components, safety and technological functions of the steam and power conversion system specified in Section "3.10.2.4 Requirements for Other Equipment of Steam and Power Conversion System in the Preliminary Design Concept" creates preconditions for meeting the requirements laid down in the IAEA document SSR 2/1 [L. 252] Req. 77, 6.57 and Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276] (124).

3.10.3 SUMMARY PRELIMINARY ASSESSMENT OF STEAM AND POWER CONVERSION SYSTEM DESIGN CONCEPT

The ETE3,4 design shall meet the safety requirements specified in Section "3.3.1.1 Basic Legislative Requirements for Safety" as well as the requirements specified in Sections 3.10.1.1 to 3.10.1.4, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and its implementing decrees for nuclear safety, radiation protection, and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270].

To meet the above specified requirements, the ETE3,4 design shall use systems, structures, and components and their equipment integrated within the steam and power conversion system, which shall ensure, with adequate reliability and resistance, qualitatively and quantitatively adequate technological functions, in accordance with the specified requirements for the prescribed safety functions.

The specification of the design described in Section "3.10.2 Design Characteristics for the Needs of the Preliminary Assessment", was created based on the requirements that the applicant for licence imposed on the potential suppliers of the nuclear installation within the tender procedure, and it makes up the concept of the design for this part of the design. The partial assessments demonstrate that the expected design of the steam and power conversion system creates preconditions for compliance with the relevant requirements for systems, structures and components, and safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276], IAEA document SSR 2/1 [L. 252] and WENRA document [L. 27].

The particular method of technical implementation of the individual requirements for the functions of the steam and power conversion system described in Section 3.10.2 shall be specified in detail in the nuclear power plant design.

3.11 RADIOACTIVE WASTE MANAGEMENT

This comprehensive part of the Initial Safety Analysis Report is structured into three basic sections:

The introductory Section 3.11.1 summarizes, analyzes, and specifies basic legislative requirements in the area of radioactive waste management, focusing on the specification of their sources. This section contains the description and stipulation of the basic requirements for the system of identification, description and methodology of the quantification of source terms and functions of the radioactive waste management systems, which comprise an envelope of parameters and design requirements for these systems. The following part of this section contains the requirements for the solid, liquid and gaseous waste management systems and the operating media and discharge radiation monitoring systems.

The following Section 3.11.2 contains description and specification of the basic requirements for the functions of the radioactive waste management system in the form of "envelope-applied" design requirements for relevant process subsystems, specifically within all the relevant designs taken into account in the tender procedure in progress. The objective of this section is to formulate the general characteristics of the design for the purposes of partial preliminary assessment.

The final Section 3.11.3 contains a summary preliminary assessement of the design concept of the radioactive waste management system, summarizing the conclusions of the partial preliminary assessments presented in Section 3.11.2. The assessment of the summarized requirements contains a preliminary design concept assessment required by the law.

At the next stage of the licensing documents the licence applicant shall provide evaluating and supporting information, allowing assessment of future operator's ability to handle, collect, sort, store, process and manage liquid, gaseous and solid waste that may contain radioactive substances. The section shall also be supplemented with more detailed information at the depth and in the structure described in RG 1.206 [L. 275].

3.11.1 BASIC LEGISLATIVE REQUIREMENTS FOR RADIOACTIVE WASTE MANAGEMENT

3.11.1.1 SOURCE TERMS

Radioactive waste (RAW) is generated by nuclear power plant operation due to leakage of fission products from the fuel or due to neutron activation of materials and media in the reactor core or spent fuel storage pool. In addition, the fission and activation radionuclides of the reactor coolant contaminate various gaseous, liquid media and solid materials.

For the activity of liquid media, the primary source term is the water from the primary circuit, where radionuclides are present both in soluble and insoluble form (mainly the corrosion products).

In principle, gaseous radioactive waste is the air mass from active process circuits or from rooms within the controlled area of the nuclear power plant (monitored), contaminated radioactive gases and aerosols, the expected activity of which does not allow uncontrolled release to the outer atmosphere. The largest source of the activity of gaseous radioactive waste (mainly gaseous fission products) is the ventilation of the primary circuit water deaerator.

Solid radioactive waste is generated depending on the reactor operating mode, mainly during regular outages, maintenance and cleaning works, decontamination of equipment and rooms, etc. The contact with active media, mainly the NPP primary circuit water, often the mediated contamination deposit on the equipment, is the source of contamination of various objects (clothes, protective equipment, defective further unusable parts of the equipment, etc.). In addition to this randomly or irregularly generated radioactive waste, a regular generation of solid radioactive waste may be expected, i.e. filters of active ventilation systems, activated measuring sensors and surveillance specimen capsules.

This section contains the basic legislative requirements for the identification, description and methodology of the quantification of source terms. The related licensing documents shall contain the methodology and calculations of source terms, such as fission products, corrosion products, coolant activation, fuel impurity activation, tritium and transuranium production, air activation in reactor cavity, storage pool water activity and secondary circuit activity.

Requirements Requesting Identification of Source Terms:

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (27), (127), IAEA SSR 2/1 2.9, 4.4, 5.71, 6.69

A safety analysis shall be performed, identifying all sources of radioactivity and demonstrating that the radiation doses that could be received by radiation personnel and/or individuals from the population due to nuclear installation operation and the possible environmental impact are in compliance with the requirements laid down by a special legal decree (Decree No. 307/2002 Coll. [L. 4]) and are at the lowest reasonably achievable level.

Conservative source terms shall be described, i.e. a conservative assumption of the quantity and composition of radioactive substances in the reactor coolant system, in the spent fuel storage system, in the secondary circuit and in the air mass in the vicinity of such systems, used in designing the coolant purification systems and the related radioactive waste processing systems. At the same time, realistic source terms shall be described, i.e. a realistic assumption of the quantity and composition of radioactive substances in the reactor coolant system, in the spent fuel storage system, in the secondary circuit and in the air mass, used in designing the authorized limits for discharges.

Reference: Decree No. 195/1999 Coll., Section 4(3)

The quality and suitability of computational programs used for the analyses important for nuclear safety shall be verified.

3.11.1.2 LIQUID WASTE MANAGEMENT SYSTEMS

The objective of the processing of contaminated liquid media is to concentrate the activity to the least volume possible. That way, the relatively small amount of the medium that can be labelled as radioactive waste (RAW) is generated on one hand, as well as the relatively large amount of the decontaminated medium for further use, processing or discharge on the other hand.

The proven methods of mechanical and ion-exchange filtration, sorption, sedimentation, centrifugation and vaporisation are used as the methods for radioactive water processing.

The processing and treatment of concentrated liquid radioactive media is provided by their fixation to a matrix.

This section contains the basic legislative requirements for liquid waste management system design. The section in the Preliminary Safety Analysis Report, which shall follow, shall contain the description of the design basis of the liquid waste management system, description of the system and evaluation of discharges associated with liquid waste.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (131)

The nuclear installation must be equipped with means enabling liquid waste discharge control and radioactive waste management including sufficient retention and storage areas.

Reference: IAEA SSR 2/1, Req. 78

Plant shall be provided for treating solid radioactive waste and liquid radioactive waste at the nuclear power plant to keep the amounts and concentrations of radioactive releases below the authorized limits on discharges and as low as reasonably achievable.

Reference: IAEA SSR 2/1, 6.59

Systems and facilities shall be provided for the management and storage of radioactive waste on the nuclear power plant site for a period of time consistent with the availability of the relevant disposal option.

Reference: IAEA SSR 2/1, 6.60

The design of the plant shall incorporate appropriate features to facilitate the movement, transport and handling of radioactive waste. Consideration shall be given to the provision of access to facilities and to capabilities for lifting and for packaging.

Reference: IAEA SSR 2/1, Req. 79

The plant shall be provided for treating liquid radioactive effluents to keep their amounts below the authorized limits on discharges and as low as reasonably achievable.

Reference: IAEA SSR 2/1, 6.61

Liquid radioactive effluents shall be treated at the plant so that exposure of members of the public due to discharges to the environment is as low as reasonably achievable.

Reference: IAEA SSR 2/1, 6.62

The design of the plant shall incorporate suitable means to keep the release of radioactive liquids to the environment as low as reasonably achievable and to ensure that radioactive releases remain below the authorized limits on discharges.

3.11.1.3 GASEOUS WASTE MANAGEMENT SYSTEMS

The processing of gaseous radioactive waste involves the separation of radioactive substances from the contaminated air mass by filtration or their retention on suitable adsorption materials for the period of time significant in terms of reduction in their activity depending on the half-life and, if appropriate, by dissolving the remaining activities to the level acceptable for its discharge to the environment.

Gaseous radioactive waste is mainly composed of:

- Noble gas radionuclides
- Tritium
- Radioactive aerosols
- Radionuclides of iodine and other halogens

This section contains the basic legislative requirements for gaseous waste management system design. The section in the Preliminary Safety Analysis Report, which shall follow, shall contain the description of the design basis of the gaseous waste management system, description of the system and evaluation of discharges associated with gaseous waste.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (131)

The nuclear installation must be equipped with means enabling gaseous waste discharge control and radioactive waste management including sufficient retention and storage areas.

Reference: IAEA SSR 2/1, Req. 79

The plant shall be provided for treating gaseous radioactive effluents to keep their amounts below the authorized limits on discharges and as low as reasonably achievable.

Reference: IAEA SSR 2/1, 6.61

Gaseous radioactive effluents shall be treated at the plant so that exposure of members of the public due to discharges to the environment is as low as reasonably achievable.

Reference: IAEA SSR 2/1, 6.63

The cleanup equipment for the gaseous radioactive substances shall provide the necessary retention factor to keep radioactive releases below the authorized limits on discharges. Filter systems shall be designed so that their efficiency can be tested, their performance and function can be regularly monitored over their service life, and filter cartridges can be replaced while maintaining the throughput of air.

3.11.1.4 SOLID WASTE MANAGEMENT SYSTEMS

The objective of the solid radioactive waste management is to minimize the volume of solid radioactive waste and treat it in compliance with the acceptance criteria for their disposal in the radioactive waste repository. The solid radioactive waste management is composed of a number of linked steps, which include the controlled collection and sorting (by waste type and activity) in the point of origin, removal to the central workplace, sorting by activity and type (method of treatment and storage), waste processing (fragmentation, volume reduction), storage, treatment, transport and disposal.

This section contains the basic legislative requirements for solid waste management system design. The section in the Preliminary Safety Analysis Report, which shall follow, shall contain the description of the design basis of the solid waste management system, description of the solid waste processing and treatment systems and evaluation of capacities and balances of solid waste.

Reference: IAEA SSR 2/1, Req. 78

The plant shall be provided for treating solid radioactive waste to keep the amounts of radioactive waste as low as reasonably achievable.

Reference: IAEA SSR 2/1, 6.59

Systems and facilities shall be provided for the management and storage of radioactive waste on the nuclear power plant site for a period of time consistent with the availability of the relevant disposal option.

Reference: IAEA SSR 2/1, 6.60

The design of the plant shall incorporate appropriate features to facilitate the movement, transport and handling of radioactive waste. Consideration shall be given to the provision of access to facilities and to capabilities for lifting and for packaging.

3.11.1.5 PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS

This section contains the basic legislative requirements for design of the process and effluent radiation monitoring system . The section in the Preliminary Safety Analysis Report, which shall follow, shall contain the description of the design basis of the process and effluent radiation monitoring system, description of the systems and characteristics of the approach to monitoring and sampling of discharges and process fluids.

3.11.1.5.1 Description of the Process and Effluent Radiation Monitoring Systems

The objective of the process and effluent radiation monitoring system is to systematically and continuously monitor the level of the radiation situation of the environment, monitor the level, movement and accumulation of activities in process circuits, systems and equipment, monitor the function and tightness of the equipment (monitoring of barriers against the release of radioactive substances) and monitor the level of gaseous and liquid radioactive waste discharges.

The process and effluent radiation monitoring systems are closely associated with the systems of work environment radiation monitoring, personal radiation monitoring, radiation monitoring of items and objects transported over the boundary of the controlled area and the restricted area, waste and process equipment monitoring, and with the radiation monitoring systems of the surrounding environment. The legislative requirements for the radiation protection system are defined in Section 3.12.1 and the concept of the design in this area is described in Section 3.12.2.

3.11.1.5.2 Requirements for the Extent of Controls Performed by Process and Effluent Radiation Monitoring Systems

Reference: Decree No. 195/1999 Coll., Section 43

The ionising radiation and radionuclide monitoring shall be provided in compliance with the specific legal decree (Decree No. 307/2002 Coll. [L. 4]).

Reference: Decree No. 195/1999 Coll., Section 45

The monitoring of gaseous and liquid radioactive substance discharges to the environment shall be provided by suitable means and shall be in compliance with the specific legal decree (Decree No. 307/2002 Coll. [L. 4]).

Safety Guide of the Sate Office for Nuclear Safety BN-JB-1.0 (119)

The nuclear installation shall have adequate means for performance and functional testing and monitoring of the equipment designed for radioactive waste treatment.

Reference: IAEA SSR 2/1 Req. 82, 6.80, 6.81, Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (129, 130, 131)

The nuclear power plant shall be equipped with adequate means and instrumentation that shall be sufficiently sensitive, easily available, type-approved, and the specified meters shall be verified and easy to verifyand with laboratories to ensure suitable radiation monitoring in operational states, during design basis accidents and, to the maximum possible extent, during design extension conditions.

Monitoring by means of stable equipment or sampling with subsequent laboratory analysis shall be used on an as needed basis for early identification of the concentration of the selected radionuclides in process circuits with liquid and gaseous media or in the environment, in normal and abnormal operational states and during accident conditions.

The stable equipment shall be used for the monitoring of radioactive and potentially radioactive waste water, before or in the course of the release to the environment.

Remark:
The legislative requirements for the liquid waste management systems are defined in Section 3.11.1.2 and the concept of the design in this area is described in Section 3.11.2.2.

The legislative requirements for the gaseous waste management systems are defined in Section 3.11.1.3 and the concept of the design in this area is described in Section 3.11.2.3.

The legislative requirements for the solid waste management systems are defined in Section 3.11.1.4 and the concept of the design in this area is described in Section 3.11.2.4.

Reference: Decree No .195/1999 Coll., Section 28(b), Section 25(2), Section 26(b), Section 47(i), Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (70, 81, 121)

The process and effluent radiation monitoring systems shall provide information necessary for the monitoring of the integrity of the barriers (e.g. primary circuit, spent fuel pool, emergency core cooling system).

Reference:Decree No. 195/1999 Coll., Section 42(1)

The presence of radionuclides in the containment that could enter the containment in accident conditions shall be monitored.

Reference: Decree No. 195/1999 Coll., Section 47(i)

The process and effluent radiation monitoring systems shall provide information on the content of radionuclides in any water, which the irradiated fuel shall be stored in or which shall be used for handling thereof.

3.11.1.5.3 Requirements for Information Systems of Process and Effluent Radiation Monitoring

The information systems of the process and effluent radiation monitoring classified as important to safety shall meet the basic requirements defined in Section "3.7.1.5.2 Requirements for Information Systems Important to Safety". The systems not classified as important to safety shall meet the basic requirements defined in Section "3.7.1.7.2 Requirements for Control Systems Not Required for Nuclear Safety Assurance". The information systems shall also meet the requirements specified in Section "3.18.1 Basic Requirements for Engineering Psychology and Ergonomics".

Reference: Decree No. 195/1999 Coll., Section 16(1, 3), Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (83, 84, 95)

The nuclear power plant shall be equipped with the information systems allowing to monitor, measure and record the operating parameters essential for assuring nuclear safety during normal, abnormal operation and in accident conditions.

Communicators of the status of parameters and components and equipment controllers shall be designed and arranged to respect human factor and ergonomic requirements for human-machine interface and to enable the operating personnel to have always enough easy to manage comprehensive information on the operation of the nuclear installation and on the results of automatic interventions of control systems. This shall enable the operating personnel to intervene, if necessary.

The control and information systems shall provide the necessary visual and audio warnings informing of new operational states and processes that deviate from the limits for normal operation and may affect nuclear safety.

The nuclear installation shall be equipped with appropriately qualified information systems capable of providing, recording and processing, on an as needed basis and in accident conditions, the following:

- Information regarding the immediate state of the nuclear installation, based on which the protective measures can be implemented, especially on the parameters and state of the systems that can affect the progress of fission reaction, reactor core integrity, integrity of the primary circuit and protective envelope and the systems associated therewith
- Basic information on the course of the accident and its recording
- Information that enables the prognosis of the spread of radionuclides and ionising radiation to the vicinity of the nuclear installation so that timely measures to protect the population could be implemented

For the accident conditions of severe accident, the design shall provide for adequate instrumentation that can be used in the conditions of such accidents and allows for the monitoring of parameters required for such accident management.

Reference: IAEA SSR 2/1 Req. 59, 6.31

There shall be instrumentation available within the radiation monitoring for measurement of the main variables, which can affect the fission processes, reactor core integrity, reactor and containment cooling systems, and means for obtaining information on the power plant necessary for its reliable and safe operation, means for identifying the state of the power plant in accident conditions as well as means for identifying the information important to accident management.

The instrumentation and equipment for information recording and storage shall be available to ensure the availability of important information on the monitoring of accident development and state of important equipment, information required for the prediction of the place of occurrence and amount of radioactive material that could escape from the areas defined by the design, and for post-accident analyses.

Reference: WENRA App. E 10.2, Issue J 2.1

The instrumentation of the process and effluent radiation monitoring systems shall be suitable for the parameters being measured and shall be qualified for operation in the specified environment and in power plant states taken into consideration.

The information containing experience from normal and abnormal operation and other information related to safety shall be sorted, documented and stored in such a manner to enable its simple reuse, search, display and evaluation by appointed personnel.

3.11.1.5.4 Requirements for Process and Post-Accident Sampling

Reference: IAEA SSR 2/1 Req. 71, 6.47

The process and post-accident sampling systems shall be available for the identification of the time development of concentrations of the selected radionuclides in fluid process systems and in gaseous and liquid samples taken from the systems or from the environment, in normal and abnormal operational states as well as in accident conditions.

Suitable means for activity monitoring in the systems containing process fluids, which could experience significant contamination, and means for the collection of process samples shall be also available.

The legislative requirements for the process and post-accident sampling systems are defined in Section "3.9.1.3.2 Process sampling systems and Post-Accident Sampling Systems" and the concept of the design in this area is described in Section "3.9.2.3.2 Process sampling systems and Post-Accident Sampling Systems in the Preliminary Design Concept".

3.11.2 CHARACTERISTICS OF RADIOACTIVE WASTE MANAGEMENT DESIGN FOR THE NEEDS OF THE PRELIMINARY ASSESSMENT

In accordance with Act No. 18/1997 Coll. [L. 2], radioactive waste (RAW) is defined as "substances, objects or equipment containing or contaminated with radionuclides, for which no further use is foreseen". In accordance with Decree No. 307/2002 Coll. [L. 4], the radioactive wastes are divided into gaseous, liquid and solid. Solid radioactive waste is classified into three basic categories, specifically to transitional, low- and medium-level and high-level waste.

The radioactive waste management system shall be linked to the system of identification, description and methodology of quantification of source terms. The requirements for this system arising from the agreed legislative sources for the preparation of the Initial Safety Analysis Report are laid down in Section "3.11.1 Basic Requirements for Radioactive Waste Management".

The radioactive waste management system shall provide for the collection, sorting, processing and treatment of all types of wastes that will be generated in the controlled area. The radioactive waste processing system shall also ensure the handling of waste and its release to the environment (under specified conditions) and to the radioactive waste repository (RAWR).

The radioactive waste processing systems shall be equipped with modern technologies providing the maximum reduction of waste to be reposed possible, providing the suitable physical characteristics of the substances released into the environment, as well as the minimum radiation dose to the personnel.

The radioactive waste management systems shall be designed so as to ensure meeting of the basic safety requirements and principles specified in Section 3.3.1. These include, in particular, the principle of defence in depth and the acceptance criteria specified for the systems, structures and components specified within the design basis, including the relevant design requirements based on the conditions of the site as specified in Chapter 2 and summarised for the most important ones in Section 2.10.

In order to be minimized, the wastes shall be separated at the point of origin depending on their activity to the active, and potentially inactive waste. The generated waste will further be separated based on the expected way of processing and treatment. The active waste will be processed using the installed technological systems. These systems shall have sufficient processing as well as storage capacity. During the whole radioactive waste management process (processing, storage, final treatment) the monitoring of characteristic values shall be ensured.

Processing of contaminated liquid media shall be driven by the effort to concentrate the activity to the least volume possible. That way, the relatively small amount of the medium that can be classified as radioactive waste is generated on one hand, and on the other hand a relatively large amount of decontaminated medium for further use. The methods for processing and final treatment of radioactive water shall include the proven methods, like filtration, centrifugation, vaporisation, drying and fixation to a matrix.

When processing gaseous RAW, the radioactive substances will be separated from contaminated air mass using filtration. The systems will be equipped in order to prevent the leakage of radionuclides into the environment (retardant absorbers, active filters, equalising tanks). When processing solid waste, the proven methods including separation, fragmentation, pressing, shall be used.

After the final treatment, the treated short-term low-level and intermediate-level radioactive waste shall be removed to the radioactive waste repository. The repository is designed not only for the reposition of the operating waste, but also the waste from the period of decommissioning of ETE3,4.

High-level waste that cannot be reposed in the radioactive waste repository, is stored in an organized way in the storage facilities of the power plant.

Radioactive Waste Processing System

The system shall provide radioactive waste processing in gaseous, liquid and solid forms.

The gaseous waste is generated mainly from the continuous removal of gases generated by radiolysis in the reactor or generated as gaseous fission products from the coolant. The gaseous waste shall be stripped of dust and moisture on dust filters and subsequently stripped of radioactive aerosols on adsorption filters, thus converting it to solid or liquid form. The purified gases shall be further stored in fuel digestion tanks where their activity is reduced by the way of natural decay, and after verification they shall be released through the ventilation stack in a controlled way.

The requirements for the gaseous waste management systems arising from the agreed legislative sources for the preparation of the Initial Safety Analysis Report are laid down in Section "3.11.1.3 Gaseous Waste Management Systems" and the concept of the design in this area is described in Section "3.11.2.3 Gaseous Waste Management Systems in the Preliminary Design Concept".

Liquid waste is mainly the result of the core circuit coolant purification. The coolant shall be stripped of the impurities on mechanical filters and ion exchangers and the resultant radioactive waste shall be subsequently concentrated in vaporisers. After treatment the majority of the coolant and a part of the chemicals shall be re-used in the primary circuit and the rest shall be released to the water flow after controlled verification. The ion exchangers and concentrated vaporiser waste shall be converted to the solid form using fixation to a different material (most frequently cement, bitumen or glass).

The requirements for the liquid waste management systems arising from the agreed legislative sources for the preparation of the Initial Safety Analysis Report are laid down in Section "3.11.1.2 Liquid Waste Management Systems" and the concept of the design in this area is described in Section "3.11.2.2 Liquid Waste Management Systems in the Preliminary Design Concept".

The solid waste shall be separated, possibly fragmented, and stored in steel barrels. The solidified and solid waste shall be transported in steel barrels to the radioactive waste repository.

The requirements for the solid waste management systems arising from the agreed legislative sources for the preparation of the Initial Safety Analysis Report are laid down in Section "3.11.1.4 Solid Waste Management Systems" and the concept of the design in this area is described in Section "3.11.2.4 Solid Waste Management Systems in the Preliminary Design Concept".

Monitoring of Discharges

The discharge monitoring system shall be used to monitor radioactive substances discharged into the air or watercourses from the power plant and shall provide control for not exceeding the authorized limits of discharges set by the State Office for Nuclear Safety and, in the case of liquid waste discharges, also by the Regional South Bohemia Authority.

The gaseous waste discharges shall be monitored in order to control the observance of the specified legislative and authorized limits and signaling of an excess of reference levels for any leakage of radioactive substances to the environment.

The control of gaseous waste discharge monitoring shall include:

- Control of the observance of the specified limits for radioactive substance discharges to the atmosphere during operation under normal conditions as well as accident and post-accident situations
- Signaling of radioactive substance leakage into the environment and specification of the amount of the activity leaked into the environment through the ventilation stacks under emergency conditions

Balance (off-line) monitoring of gaseous waste discharges shall be provided based on the sampling and subsequent spectrometric evaluation. The system shall include the gamma-spectrometric measurements of the samples of aerosols, iodine, inert gases, carbon and tritium. Representative samples shall be taken by relevant sampling devices that shall be subjected to regular calibrations and examinations to be carried out by an authorized metrological centre.

Liquid waste discharges shall be monitored in order to control the observance of the specified limits and signaling of liquid radioactive substance leakage. If the licensed activity of liquid waste discharges from the selected control reservoirs is exceeded, the system shall provide for the interruption of their release.

The control of liquid waste discharges shall be provided using continuous and discontinuous monitoring of the activity of waters drained from the premises of Temelín NPP and shall be used to provide for the balancing of discharged radioactive substances. The control shall also be meant to prevent undesirable leakage of radioactive substances into the environment, to signal the exceeding of set reference levels and to provide interruption of the discharge upon achieving the intervention levels when emptying individual control reservoirs.

The requirements for the media and discharge radiation monitoring systems arising from the agreed legislative sources for the preparation of the Initial Safety Analysis Report are laid down in Section "3.11.1.5 Operating Media and Discharge Radiation Monitoring Systems".

3.11.2.1 SOURCE TERMS IN PRELIMINARY DESIGN CONCEPT

An analysis of source terms in all normal operational states, including impacts of maintenance and testing, as well as in all abnormal operational states shall be performed for the purposes of designing the radioactive waste management systems.

The used theoretical models shall be based on the recognized scientific fundamentals and should be sufficiently verified against suitable experimental data.

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for the source terms specified in Section "3.11.2.1 Source Terms in Preliminary Design Concept" creates preconditions for meeting the requirements laid down in Decree No. 195/1999 Coll. [L. 266] Section 4(3), IAEA document SSR 2/1 [L. 252] Req. 2.9, 4.4, 5.71, 6.69 and Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276] (27), (127).

3.11.2.2 LIQUID WASTE MANAGEMENT SYSTEMS IN PRELIMINARY DESIGN CONCEPT

The liquid radioactive waste processing system shall be capable of receiving, segregating, processing, releasing or recycling liquid waste in the course of all normal operating conditions. In addition, the system shall transport concentrates, ion exchangers and waste from filter washing to the solid waste processing system.

The system shall have the capability of taking the maximum expected volume of generated liquid waste, The system shall allow using the alternative or backup procedures for waste processing, which will ensure power plant availability and continuous receipt of liquid waste.

Liquid radioactive waste shall be segregated on its initial intake into the following categories:

- Equipment drainage
- Waste containing boric acid
- Drainage from cubicle floors
- Chemical waste
- Detergent waste (e.g. from laundry)
- Mixed waste (e.g. contaminated oil)

Two parallel collecting tanks shall be installed where large inlets are expected.

Each subsystem shall be equipped with own sampling tanks that shall be used for taking of samples, which shall be analyzed before release or recycling.

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for the liquid waste management systems specified in Section "3.11.2.2 Liquid Waste Management Systems in Preliminary Design Concept" creates preconditions for meeting the requirements laid down in the IAEA document SSR 2/1 [L. 252] Req. 78, 6.59, 6.60, Req. 79, 6.61, 6.62 and Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276] (131).

3.11.2.3 GASEOUS WASTE MANAGEMENT SYSTEMS IN PRELIMINARY DESIGN CONCEPT

The gaseous radioactive waste processing system shall be in operation in all operational states of the power plant.

The system shall be designed to prevent hydrogen explosion and to maintain the system integrity even in the case of the formation of an explosive mixture of hydrogen and oxygen. Gases in the system shall be permanently maintained as non-flammable by means of oxygen-free atmosphere.

The system shall process radioactive gases formed in the reactor, which are released from the bleed and feed system, and from the boric acid processing system. The points of generation of gaseous radioactive waste and the flows from individual sources shall be identified.

The system shall be designed to:

- Be capable of processing gases given by the combination of all sources
- Meet the limits for discharge
- Release gaseous discharge after "extinction" of radioactive substances in the retention tanks or activated carbon filters through the monitoring equipment in the ventilation systems
- Be capable of continuous and periodic operation within the flows, as required by the bleed and feed system or the boric acid processing system

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for the gaseous waste management systems specified in Section "3.11.2.3 Gaseous Waste Management Systems in Preliminary Design Concept" creates preconditions for meeting the requirements laid down in the IAEA document SSR 2/1 [L. 252] Req. 79, 6.61, 6.63 and Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276] (131).

3.11.2.4 SOLID WASTE MANAGEMENT SYSTEMS IN PRELIMINARY DESIGN CONCEPT

The solid radioactive waste management system shall be designed to prevent radioactive material from escaping to the environment as well as to the areas of the power plant that could cause health risks and endanger the safety of the public and operating personnel.

The selection of the waste processing method for each step of the process shall be based on the assessment of the type and amount of the generated waste.

A press shall be installed for volume reduction of solid compressible waste that shall be suitable for compression into 200 l barrels.

The radioactive waste solidification technology shall be based on the use of the inorganic-base matrix, into which radioactive waste shall be added.

Solid waste that shall be the final product of the process of volume reduction, solidification and packaging shall meet the relevant requirements of the Czech legislation and the requirements for disposal in the radioactive waste repository in Dukovany.

The temporary waste repository on the premises of the power plant shall be dimensioned for the minimum of 5 years, taking into account the expected normal operation.

The temporary repository shall meet the following requirements:

- It shall be designed as an inner space
- It shall be provided with a "reloading point" for the use of mobile transportation means (e.g. railway car, truck, etc.)
- Sufficient shielding appropriate to the situation in the individual areas of the repository shall be used. Separated areas for low- and high-level containers shall be available.
- Fire alarm system and fire-extinguishing system shall be used
- Means against unexpected damage to the container shall be used, including monitoring and control system in the case of air contamination

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for the solid waste management systems specified in Section "3.11.2.4 Solid Waste Management Systems in Preliminary Design Concept" creates preconditions for meeting the requirements laid down in the IAEA document SSR 2/1 [L. 252] Req. 75, 6.59, 6.60.

3.11.2.5 PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS IN PRELIMINARY DESIGN CONCEPT

The process and effluent radiation monitoring systems shall ensure:

- Monitoring of barriers integrity,
- Monitoring of radioactive effluents,
- Personnel exposure risk warning,

The process and effluent radiation monitoring systems shall generate warning messages or initiate measures to reduce the leak of radionuclides after reaching the pre-set values.

The relevant data provided by such systems shall be displayed at the relevant workplaces and its long-term archiving shall be ensured.

Monitoring of Barriers Integrity

The integrity of the first as well as the second barrier shall be monitored in all design conditions.

The monitoring of the environment of those buildings, the atmosphere of which could be exposed to the release of radionuclides, shall be ensured.

The monitoring of the coolant in the component cooling water system and secondary side of the steam generators shall be ensured.

The monitoring of radionuclides in the atmosphere of the containment shall be ensured during post-accident conditions.

The process and effluent radiation monitoring systems shall be available to monitor the leaks of the activity to the secondary circuit. These systems shall measure the activity in the main steam lines, steam generator blowdown and in the condenser.

The process and effluent radiation monitoring systems shall monitor the contamination of air mass in the main ventilation lines from the areas exposed to contamination.

Monitoring of Radioactive Effluents

The process and effluent radiation monitoring systems shall ensure the monitoring of liquid radioactive effluents. After exceeding the pre-set value, the discharge line shall be closed and the warning message shall be signalled.

The process and effluent radiation monitoring systems shall ensure the monitoring of gaseous effluents from the nuclear installation.

3.11.3 SUMMARY PRELIMINARY ASSESSMENT OF THE CONCEPT OF RADIOACTIVE WASTE MANAGEMENT DESIGN

The NPP3, 4 design shall meet the safety requirements specified in Section 3.3.1.1 "Basic legislative requirements for provision of safety", as well as the requirements specified in Sections 3.11.1.1 to 3.11.1.5, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and its implementing decrees regarding the nuclear safety, radiation protection, and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270] documents.

To meet the above specified requirements, the NPP3, 4 design shall use systems, structures, and components and their equipment integrated within the radioactive waste management systems, which shall ensure, with adequate reliability and resistance, qualitatively and quantitatively adequate technological functions, in accordance with the specified requirements for the prescribed safety functions.

The specification of the design described in Section "3.11.2 Design Characteristics for the Needs of the Preliminary Assessment" was created based on the requirements that the applicant for licence imposed on the potential suppliers of the nuclear installation within the tender procedure, and it makes up the concept of the design for this part of the design. The partial assessments demonstrate that the expected design of the radioactive waste management systems creates preconditions for compliance with the relevant requirements for systems, structures and components, and safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276], IAEA document SSR 2/1 [L. 252] and WENRA document [L. 27].

The particular method of technical implementation of the individual requirements for the functions of the radioactive waste management systems described in Section 3.11.2 shall be specified in detail in the nuclear power plant design.

3.12 RADIATION PROTECTION

This comprehensive part of the Initial Safety Analysis Report is structured into three basic sections:

The introductory Section 3.12.1 summarizes, analyzes, and specifies basic legislative requirements in the area of radiation protection, focusing on the application of the ALARA principle, identification of ionising radiation sources in the future power plant, radiation protection design, evaluation of doses and monitoring for the purposes of radiation protection.

The following Section 3.12.2 contains the description and specification of the basic requirements for the radiation protection specified in the previous section, which comprise an envelope of parameters and design requirements for these systems, within the framework of all designs taken into consideration in the tender procedure in progress. The objective of this section is to formulate the general characteristics of the design for the purposes of partial preliminary assessment.

The final Section 3.12.3 contains a summary preliminary assessement of the concept of the radiation protection design, summarizing the conclusions of the partial preliminary assessments presented in Section 3.12.2. The assessment of the summarized requirements contains a preliminary design concept assessment required by the law.

At the next stage of the licensing documents, the applicant shall provide evaluating and supporting information that will make it possible to assess whether or not the radiation protection principles defined by Act No. 18/1997 Coll. [L. 2], in particular Section 4 and the relevant implementing decrees are observed in carrying out activities associated with the utilisation of nuclear energy, activities leading to exposure and interventions to reduce the exposure. The section shall also be supplemented with more detailed information at the depth and in the structure described in RG 1.206 [L. 275].

3.12.1 BASIC LEGISLATIVE REQUIREMENTS FOR RADIATION PROTECTION

3.12.1.1 PERSONNEL EXPOSURE OPTIMISATION (ALARA PRINCIPLE)

This part of the Initial Safety Analysis Report summarizes the basic national and international design requirements for the radiation protection system. At the next stage of the licensing documents, the section shall be complemented with an evidence of the capability of the radiation protection system to meet the safety functions and shall contain the description of the key measures to ensure the optimisation of personnel exposure (ALARA principle) in terms of the determination of concepts, design requirements and operational aspects at the depth of detail and in the structure in accordance with RG 1.206 [L. 275].

Reference: Decree No. 195/1999 Coll., Section 5, Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (136), IAEA SSR 2/1 4.4

The nuclear installation shall be provided with the radiation protection in its structures and in its vicinity in accordance with the special legal decree (Decree No. 307/2002 Coll. [L. 4]). To meet the specified requirements, the design shall create the appropriate conditions.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (127)

In all operational states of the nuclear installation and at all planned discharges of radioactive substances all sources of ionising radiation shall be subject to administrative and technical control and exposures shall be kept below the exposure limits, possibly authorized limits at as low as reasonably achievable level.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (132)

The structural materials and operating media shall be selected with regard to the formation of activation and corrosion products, the type and amount of which together with radioactive waste shall be justified and shall be at as low as reasonably achievable level.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (133), IAEA SSR 2/1 3.6 (1)

The areas of the nuclear installation must be designed to keep the radiation risk at as low as reasonably achievable level. The areas must be classified and categorized with regard to this risk and barriers and shielding preventing radionuclides from spreading and contamination and taking into account the places and time required for works that should be carried out during normal and abnormal operation as well as under accident conditions.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (134)

The equipment requiring frequent operation or maintenance shall be preferably located in the areas of low exposures.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (135)

The sufficient number and capacity of the appropriately equipped points for contamination measurement and decontamination of personnel, nuclear installation and its equipment shall be available.

3.12.1.2 SOURCES OF IONISING RADIATION

This part of the Initial Safety Analysis Report summarizes the basic national and international design requirements for the identification of ionising radiation sources in the future power plant. At the following stage of the licensing process, the section shall contain the identification of ionising radiation sources, focusing on the characteristics of internal sources and particularly sources of airborne radioactive materials. At the next stage of the licensing documents, this section shall be completed with exact identification of ionising radiation sources at the depth of detail and in the structure specified in RG 1.206 [L. 275].

3.12.1.3 RADIATION PROTECTION DESIGN

This part of the Initial Safety Analysis Report contains the basic national and international design requirements for the radiation protection design of the future power plant. The following stage of the licensing documents structured in accordance with RG 1.206 shall contain a list of adopted technical measures in the designs of the radiation protection systems and in the systems ensuring radiation safety of personnel and the population in the vicinity of Temelín NPP during operation of the power plant. In addition, it shall deal with the characteristics of the approach to shielding, radiation aspects of ventilation systems and the characteristics of monitoring systems.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (128)

The design shall contain the analysis of tasks and processes in terms of radiation protection, identify all sources of ionising radiation and provide the protection against its adverse effects.

Reference: WENRA App. E 1.1

The design of the nuclear installation shall contain preventive measures to prevent leak of radioactive substances for the case of failure as well as means to mitigate the consequences of operational events and accident conditions. The nuclear installation shall be designed to protect the population and personnel against exposure exceeding the specified limits and to achieve the as low as reasonably achievable level.

Reference: IAEA SSR 2/1 4.3

The nuclear installation shall be designed:

- To virtually eliminate events that could cause an excessive exposure to persons or significant leaks of radioactive substances
- So that the events with significant probability of occurrence have no or less significant radiation consequences

Reference: IAEA SSR 2/1 Req. 81

The design shall make sure that the exposure of the population and personnel of the nuclear installation taken into consideration does not exceed the specified limits of exposure and is as low as reasonably achievable. To reduce the radiation impacts of the installation on personnel, the population and the environment, the following requirements should be met:

Reference: IAEA SSR 2/1 6.69

The sources of radiation across the power plant shall be comprehensively identified and the radiation associated therewith shall be as low as reasonably achievable. The integrity of fuel cladding shall be maintained as well as the formation and transport of corrosion and activation products.

Reference: IAEA SSR 2/1 6.70

The materials used for the manufacture of structures, systems and components shall be selected to keep the activation of materials at as low as reasonably achievable level.

Reference: IAEA SSR 2/1 6.71

The necessary preventive measures shall be taken to prevent the work environment and the environment from contaminating due to the leak and dispersion of various radioactive substances and waste.

Reference: IAEA SSR 2/1 6.72

The layout shall appropriately provide the adequate control of personnel for the entry and stay in the places with potential risk of exposure or with potential contamination with radioactive substances. This layout together with the design of the ventilation systems shall contribute to the prevention or reduction of radiation doses and contamination risk.

Reference: IAEA SSR 2/1 6.73

The nuclear installation shall be divided into zones in accordance with the expected stay of persons and in accordance with the expected radiation situation during operational states (including refuelling, maintenance and inspection of the equipment) as well as during accident conditions. The sources of ionising radiation shall be shielded to keep the exposure at as low as reasonably achievable level.

Reference: IAEA SSR 2/1 6.74

The layout of the power plant shall make sure that the radiation doses for operating personnel during operation, inspection, maintenance and refuelling are as low as reasonably achievable. The use of specific protective equipment shall be considered to meet these objectives.

Reference: IAEA SSR 2/1 6.75

The equipment requiring frequent maintenance or operation shall be located in the areas of low level of dose equivalent rate to reduce the load applied to operating personnel.

Reference: IAEA SSR 2/1 6.76

The nuclear installation shall be designed to assure decontamination of personnel, operating areas as well as individual facilities.

3.12.1.4 EVALUATION OF RADIATION DOSES

This part of the Initial Safety Analysis Report contains the basic national and international legislative requirements for the evaluation of radiation doses. The following licensing documents shall contain the evaluation of effective doses to be received by operating personnel of the nuclear power plant during normal operation and in all states assumed by the design from external sources of radiation or from receipt of radioactive substances in organism at he depth of detail and in the structure in accordance with RG 1.206 [L. 275]. The next task shall be the comparison of the specified limits and the recommendations arising from the "Basic Safety Standards". This comparison and the assessment of technical design shall be then use for the assessment of the extent of the application of ALARA principles to the design of the power plant. At the same time, this analysis shall serve as a basis for further technical and organisational measures to be adopted in the phase of preparation of the operation in order to further improve the work conditions in terms of radiation hygiene.

Reference: Decree No. 195/1999 Coll., Section 45

The discharge of radionuclides to the environment shall meet the requirements stipulated by the special legal decree (Decree No. 307/2002 Coll. [L. 4]).

3.12.1.5 MONITORING FOR THE PURPOSES OF RADIATION MONITORING

This part of the Initial Safety Analysis Report contains the basic legislative requirements for the design of the monitoring systems for the needs of radiation protection. The Preliminary Safety Analysis Report, which shall follow in the next stage of the licensing procedure for the new nuclear installation, shall provide the concept of design, design principles and the description of monitorings programmes in the areas of personal monitoring, radiation situation monitoring in work areas, monitoring of operating media, environmental monitoring and monitoring of discharges at the depth of detail and in the structure in accordance with RG 1.206 [L. 275].

The radiation monitoring of operating media and discharges is described in detail in Section "3.11.1.5 Operating Media and Discharge Radiation Monitoring Systems" and the concept of design in this area is described in Section "3.11.2.5 Operating Media and Discharge Radiation Monitoring Systems in Preliminary Design Concept".

Reference: Decree No. 195/1999 Coll., Section 43

The ionising radiation and radionuclide monitoring shall be provided in compliance with the special legal decree (Decree No. 307/2002 Coll. [L. 4]).

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (129), IAEA SSR 2/1 Req. 82, 6.77

The nuclear installation shall be equipped with the appropriate technical equipment and instrumentation (sensitive, easily available, type-approved, approved and easy to certify) and laboratories enabling to monitor and evaluate the radiation situation in operational states, design basis accidents and, as reasonably feasible, during design extension conditions including signalling of exceeding the specified limits and documentation of data.

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (130), IAEA SSR 2/1 6.79, 6.82

The technical equipment or the data obtained therefrom shall be available to operating personnel and shall include at least the following measurements:

- Dose rates by stationary instruments in rooms with increased risk of exposure, which are readily accessible to operating personnel and the access to which shall be time-limited due to expected change of the dose rate, in all operational states, while the basic monitoring of the selected points shall also be available under accident conditions,
- Surface contamination, the stationary monitors for which shall be located at exits from the controlled area to ensure the control of surface contamination of persons and material,
- Activity of gaseous radioactive substances in the air in rooms readily available to operating personnel, where the activity could achieve the values requiring the introduction of protective measures,

- Activity of gaseous and liquid radioactive substances in the systems of the nuclear installation and in discharges,
- Effective doses and contamination of persons including registration and recording of such doses

Reference: Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 (131), IAEA SSR 2/1 6.81

The nuclear installation shall be equipped with means enabling gaseous waste discharge control and radioactive waste management including sufficient retention and storage areas.

Reference: IAEA SSR 2/1 6.78

For the case of accident conditions, stable monitors shall be installed where appropriate to signal the level of ionising radiation. These instruments shall provide the necessary information together with its transmission to the control room or other suitable control positions to enable the operator to take the necessary corrective or protective measures.

Reference: IAEA SSR 2/1 6.80

The necessary stable equipment and laboratory equipment shall be available in all operational states and accident conditions for early identification of the concentrations of selected radionuclides in the systems containing operating media, in liquid and gaseous samples taken from the systems of the power plant or from the environment.

Reference: IAEA SSR 2/1 6.83

The equipment for monitoring of effective doses and contamination of persons including registration and recording of such doses shall be provided.

Reference: IAEA SSR 2/1 6.84

The monitoring of radiation effects in the vicinity of the power plant due to the operation of the nuclear installation shall be ensured by controlling volume activities, contamination, doses and dose rate, with special emphasis placed on the following:

- Paths of radionuclides to the population, including food chain
- Radiation impacts on local ecosystems
- Possibility of the accumulation of radioactive materials in the environment
- Possibility of the escape via uncontrolled paths

3.12.2 CHARACTERISTICS OF RADIATION PROTECTION DESIGN FOR THE NEEDS OF THE PRELIMINARY ASSESSMENT

Description and Basic Functions of Radiation Protection

The radiation protection shall mean a system of technical and organisational measures to reduce the exposure of natural persons and protect the environment The objective of the radiation protection is to eliminate the deterministic effects of ionising radiation and reduce the probabilities of stochastic effects to the reasonably achievable level.

As part of the Initial Safety Analysis Report, which serves as a basis for the decision on siting of the nuclear installation, it is mainly required to document the technical level of the design concept in terms of the reduction of exposure of the population and the environmental protection as well as the expected exposure of personnel.

The issues associated with organisational measures are mainly addresses in the following phases of the permitting process, relate to the current organisational chart of the applicant for the licence for operation as well as operating regulations prepared in accordance with the supplier documentation. The radiation protection and optimisation are largely dealt with in Chapter 4 "Preliminary Assessment of Impact of Operation of Proposed Installation on Personnel, the Public and the Environment". The radiation situation on site is described in Section "2.7 Radiation Situation on Site".

3.12.2.1 PERSONNEL EXPOSURE OPTIMISATION (ALARA PRINCIPLE) IN PRELIMINARY DESIGN CONCEPT

The design of the nuclear installation shall ensure that the radiation doses of personnel shall be at the lowest reasonably achievable level and always below the limits of radiation doses in accordance with Decree No. 307/2002 Coll. [L. 4].

For these purposes, the technical and administrative measures shall be implemented in the design to minimize the sources of ionising radiation and to minimize the exposure of personnel. These measures are described in Sections 3.12.2.3 to 3.12.2.5.

Radiation targets for radiation personnel (defined in Chapter 4 of the Initial Safety Analysis Report) shall be set in the design to provide the sufficient reserve to exposure limits in accordance with Decree No. 307/2002 Coll. [L. 4].

Compliance with these targets shall be verified by means of radiation dose study described in Section 3.12.2.4 Evaluation of Radiation Doses in Preliminary Design Concept.

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for the optimisation of personnel exposure in accordance with the ALARA principle specified in Section 3.12.2.1 creates preconditions for meeting the requirements laid down in Decree No. 195/1999 Coll. [L. 266], Section 5, IAEA document SSR 2/1 [L. 252] Req. 3.6 (1) and Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276] 127 a 132-135.

3.12.2.2 SOURCES OF IONISING RADIATION IN PRELIMINARY DESIGN CONCEPT

The design shall minimize the sources of ionising radiation by means of the following measures:

- High reliability of nuclear fuel with the aim to achieve the minimum level of operating leaks
- Suitable selection of materials (mainly minimisation of cobalt) and surface finish of the equipment being in contact with ionising radiation or contaminated fluids
- Suitable chemistry minimizing the formation of corrosion

- Suitable design of equipment minimizing deposition of impurities and facilitating decontamination

The radioactive sources shall be reduced by suitable selection of materials (see Section 3.4.2.5, Reactor Materials in Preliminary Design Concept), selection of chemical composition of operating liquids (see 3.11.2.5, Operating Media and Discharge Radiation Monitoring Systems in Preliminary Design Concept), suitable layout of the power plant, shielding, suitable storage of sludge, decontamination and other means that shall be described in the next stage of the licensing documents.

3.12.2.3 RADIATION PROTECTION DESIGN IN PRELIMINARY DESIGN CONCEPT

Radiation Zoning

All sources of ionising radiation shall be identified in the design. In accordance with the results, the radiation zoning shall be carried out in the design, i.e. division of all areas of the nuclear installation into clean areas and supervised area or controlled area. Zoning shall also be carried out for the purposes of post-accident entry.

Several radiation zones shall exist within the controlled area in accordance with the amount of dose rate with the aim to minimize or fully prevent the entry into high-level areas. At the same time, zoning shall be carried out in accordance with the possibility of occurrence of surface or air contamination.

Design in Terms of Radiation Protection

The equipment and equipment layout shall be designed with the aim to minimize the radiation doses of personnel. The design shall be in accordance with the radiation targets and limits for personnel exposure for all operational states of the nuclear installation as well as for accident conditions. The discharges of radionuclides to the environment shall be below the authorized limits for operational states of the power plant and are dealt with in detail in Chapter 4 of the Initial Safety Analysis Report.

In accordance with the radiation zoning, the radioactive equipment in the design shall be isolated from the inactive equipment and the radioactive equipment shall be divided into different areas with regard to its dose rate. High-level piping running through the low-active zones shall be covered with shielding.

The auxiliary systems included in the radiation protection systems shall be designed so as to ensure satisfaction of the basic safety requirements and principles specified in Section 3.3.1. These include, in particular, the principle of defence in depth and the acceptance criteria specified for the systems, structures and components specified within the design basis, including the relevant design requirements based on the conditions of the site as specified in Chapter 2 and summarised for the most important ones in Section 2.10.

High-level equipment (e.g. tanks, filters or heat exchangers, etc.) shall be placed into shielded areas to enable inspections and maintenance of other equipment (e.g. valves, pumps, fans, etc.). To the maximum possible extent, shielding shall be part of the permanent civil structures.

The equipment located in the areas with higher dose rates shall be designed to the maximum possible extent as maintenance-free.

The equipment requiring maintenance or in-service inspections shall be, if possible, located in the areas with low dose rate. At the same time, easy access and simple and fast maintenance of the equipment in question and the sufficient area for possible mobile shielding shall be provided. If this is not possible, the remote or robotic control, remote reading of measured data, etc., shall be used to the maximum possible extent.

Robotic or remotely controlled equipment shall be largely used to reduce the radiation load of personnel.

The radiation doses shall also be reduced by means of suitable cleaning and filtration systems minimizing the spread of contamination, slightly decontaminable coating and drainage for the case of the leaks of active components.

Ventilation Systems

All areas with the risk of release (escape) of radioactivity to the work environment in the form of aerosols and gases shall be equipped with special ventilation systems designed for controlled area.

These systems shall:

- Provide suitable work conditions for operating personnel and equipment including maintenance of contamination level at the as low as reasonably achievable level
- Prevent the spread of contamination, especially keeping the direction of air flow from the areas with potentially lower contamination to the areas with potentially higher contamination
- Reduce the amount of gaseous radioactive discharge at the as low as reasonably achievable level

Material Requirements

The materials of the reactor equipment with regard to minimization of radiation exposure are specified in Section "3.4.2.5 Material Requirements for the Needs of Preliminary Assessment".

The design shall specify and lay down the material requirements for the selected equipment in terms of radiation protection. The use of suitable structural materials shall be specified with regard to the formation of activation and corrosion products during subsequent operation, both of the individual components and of the installation as a whole.

The classified equipment defined in Section 2 of Act No. 18/1997 Coll. [L. 2] shall be assigned to safety classes in terms of the requirements for nuclear safety. The criteria for classification and assignment of the classified equipment to safety classes 1, 2 and 3 shall meet the relevant requirements stipulated in Decree No. 132/2008 Coll. [L. 258].

The production and supply of materials to be used for the manufacture of the equipment of the nuclear power plant (or any part thereof) shall be provided by qualified domestic and foreign organisations.

The quality of the used materials shall mainly be controlled by analysing the chemical composition, determining mechanical properties and characteristics, and detecting surface and internal defects in accordance with the standards or technical conditions for specific type of material and semi-finished product. The extent of control and criteria for the evaluation of materials and semi-finished products shall be fully and accurately specified in the technical standards, technical conditions or technical documentation for the proposed equipment, for particular conditions of operation of manufacturing equipment. For such equipment, the content of admixtures shall be declared and the content of sulphur, phosphor and subsequently other chemical elements and admixtures shall be mainly controlled. The content of sulphur and phosphor in material shall meet the requirements of the technical conditions for particular semi-finished products or the requirements stipulated by standards. For equipment being in contact with water of the secondary circuit and included in the safety class 1, both the content of sulphur and the content of phosphor in equipment material shall not exceed 0.010% in weight. For semi-finished products designed to manufacture the ring for the reactor core, the requirement for the content of other admixtures and residual elements (except for sulphur and phosphor) shall be imposed on the individual materials used in manufacturing the components.

Entrances to and Exits from Controlled Area

Each unit of the power plant shall be equipped with minimum one service entrance to the controlled area. This entrance shall be provided with technical and administrative control of the entry of persons and material, changing rooms with decontamination and sanitary equipment, facilities for the issue of protective equipment, personal dosimeters and other monitoring equipment as needed and with measurement of surface contamination of persons by means of portal monitors equipped with turnstiles.

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for the radiation protection design specified in Section 3.12.2.3 creates preconditions for meeting the requirements laid down in the IAEA document SSR 2/1 [L. 252] Req. 81, 4.3, 6.69 – 6.66, WENRA [L. 27] App. E 1.1 and Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276] 128.

3.12.2.4 EVALUATION OF RADIATION DOSES IN PRELIMINARY DESIGN CONCEPT

An evidence shall be provided in the next phase of the licensing procedure demonstrating that the proposed design meets the requirements for the radiation protection laid down by the implementing regulations. The basic part of the evidence shall include the identification dose rates and the possible level of contamination in the individual areas of the nuclear installation and the identification and description of the requirements for maintenance and in-service inspections in these areas.

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for the evaluation of radiation doses specified in Section 3.12.2.4 creates preconditions for meeting the requirements laid down in Decree No. 195/1999 Coll. [L. 266], Section 45.

3.12.2.5 MONITORING FOR THE PURPOSES OF RADIATION PROTECTION IN PRELIMINARY DESIGN CONCEPT

Compliance with the targets set for radiation doses of personnel shall be ensured by means of personal radiation monitoring as well as monitoring of workplaces.

Personal monitoring of personnel shall be performed by measuring the received dose using personal dosimeters giving an immediate indication of the dose received and the possibility of dose rate control. In addition, the surface contamination at entries to and exits from the controlled area and at the boundaries of the sanitary nodes shall be monitored. The potential internal contamination shall be monitored on the basis of the approved programme of monitoring of personal doses, for example, by means of spectrometric methods.

The received doses shall be regularly evaluated and archived in accordance with the requirements laid down in Decree No. 307/2002 Coll. [L. 4].

The workplaces shall be monitored by means of the monitoring of dose rates and contamination using stationary instruments and, if needed, by means of mobile equipment. The equipment shall be provided with automatic alarm function signalling exceeding of the set parameters.

In addition, all objects brought out of the controlled area shall be monitored by controlling the objects and persons at the boundary of the guarded area.

In accordance with Section "3.11.2.5 Operating Media Radiation Monitoring Systems in Preliminary Design Concept", the activity of operating media and discharges shall be monitored (information on preliminary composition of discharges is provided in Chapter 4) and the operational and post-accident samples shall be taken (see Section "3.9.2.3.2 Operational and Post-Accident Sampling Systems"). In accordance with the requirements defined in Decree No. 307/2002 Coll. [L. 4], the radiation impacts of power plant discharges on the vicinity shall be regularly assessed using a validated model.

The relevant radiation data shall be transmitted to the main and emergency control room, to the technical support centre and to the radiation monitoring control room, and shall be archived on a long-term basis.

In addition, the existing monitoring of the vicinity of the power plant shall be used or extended by means of the following measurements:

- Photon dose equivalent rate measurement
- Measurement of volume activities of aerosols in the air
- Measurement of volume, specific and area activities of the samples of the environment (mainly agricultural products)
- Measurement of area activities of radionuclides in atmospheric fall-outs
- Measurement of volume activities of surface and ground waters on the premises and in the vicinity of Temelín NPP

Partial Preliminary Assessment

The preliminary design concept summarizing the key requirements for the monitoring for the purposes of radiation protection specified in Section 3.12.2.5 creates preconditions for meeting the requirements laid down in Decree No. 195/1999 Coll. [L. 266], Section 43, IAEA document SSR 2/1 [L. 252] Req. 82, 6.77 – 6.84 and Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276] (129, 130 and 131).

3.12.3 SUMMARY PRELIMINARY ASSESSMENT OF THE CONCEPT OF RADIATION PROTECTION DESIGN

The radiation protection design shall meet the requirements that are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and its implementing decrees regarding the nuclear safety, radiation protection, and emergency preparedness, as well as on the requirements specified in IAEA document SSR 2/1 [L. 252], and WENRA document [L. 27] and [L. 270].

The assessment demonstrates that the assumed radiation protection design creates preconditions for meeting the relevant requirements laid down in Decree No. 195/1999 Coll. [L. 266], Safety Guide of the State Office for Nuclear Safety BN-JB-1.0 [L. 276], IAEA document SSR 2/1 [L. 252] and WENRA document [L. 27].

The specification of the requirements for the radiation protection design described in Section 3.12.2 Characteristics of Radiation Protection Design for the Needs of the Preliminary Assessment was created based on the requirements that the applicant for licence imposed on the potential suppliers of the nuclear installation within the tender procedure, and it makes up the concept of the design for this part of the design.

The particular method of implementation of the radiation protection design, as described in Section 3.12.2, shall be specified in detail in the nuclear power plant design.

3.13 OPERATION

This summary part of the ISAR in its introductory Section 3.13.1 sums up, analyses and specifies basic legislation requirements in the area of operation with the focus on specification of requirements regarding organization structure of the plant operator and the determination of policy of operation organization for training of personnel and training programmes and equipment. Further, the chapter contains requirements for planning of emergency situation activities, implementation of programmes for operation stage, requirements for operation processes, requirements for securing of physical protection and personnel competence. The following Section 3.13.2 contains specifications of basic requirements in the area of operation applied to potential contractors of the future nuclear power plant forming an envelope of parameters and design requirements in this area and is used for determination of the character of the design for the needs of preliminary assessment. Based on the assessment of the summary of requirements so obtained, the final Section 3.13.3 contains preliminary assessment of the design concept as required by the law. Within the following stage of licence documentation, the applicant shall provide evaluation and support information which shall allow to the plant operator to be able to establish and maintain technical and operation personnel in adequate numbers, qualification and competences. The Section shall also be supplemented with more detailed information with articulation according to RG 1.206 [L. 275].

3.13.1 BASIC LEGISLATION REQUIREMENTS FOR THE AREA OF OPERATION

3.13.1.1 PLANT OPERATOR'S ORGANIZATION STRUCTURE

The objective is to specify the requirements for the plant operator's organization structure, rules of implementation of changes of the requirements and determination of authority and responsibility of the personnel important to nuclear safety. Within the following stage of licence documentation, the section shall contain a description of the organization of plant operation and the responsibilities of the operating personnel providing the document of preparation and maintenance of the requisite qualification of a sufficient number of plant operating personnel to secure safe and reliable operation.

Reference: WENRA Issue B 1.1

The organisational structure for safe and reliable operation of the nuclear power plant, and for ensuring an appropriate response in emergencies, shall be justified and documented.

Reference: WENRA Issue B 1.2

The adequacy of the organisational structure shall be assessed when organisational changes are made which might be significant for safety. Such changes shall be justified in advance, carefully planned, and evaluated after implementation.

Reference: WENRA Issue B 1.3

Responsibilities, authorities, and lines of communication shall be clearly defined and documented in the organizational structure for all staff with duties important to safety.

3.13.1.2 TRAINING REQUIREMENTS

The objective is to specify requirements for determination of the policy of the operating organization for training of staff, competence and qualification of staff and training programmes and facilities. Attention shall be paid particularly to the preparation and training of workers whose activities are critical in terms of nuclear safety and radiation protection. In the following stage of licence documentation, the section shall contain the description and schedule of preparation of licensed workers including requalification and the description and schedule of preparation on unlicensed workers of the operating organization including the identification of control documentation effective in this area.

Policy

Reference: WENRA Issue D 1.1

The plant operator shall establish an overall training policy and a comprehensive training plan on the basis of long-term competency needs and training goals that acknowledges the critical role of safety. The training plan shall always be kept up-to-date.

Reference: WENRA Issue D 1.2

A systematic approach to training shall be used to provide a logical progression, from identification of the competences required for performing a job, to the development and implementation of training programmes including respective training materials for achieving these competences, and to the subsequent evaluation of this training.

Competence and qualification

Reference: WENRA Issue D 2.1

Only qualified persons that have the necessary knowledge, skills, and safety attitudes shall be allowed to carry out tasks important to safety. The plant operator shall ensure that all personnel performing safety-related duties including contractors have been adequately trained and qualified.

Reference: WENRA Issue D 2.2

The plant operator shall define and document the necessary competence requirements for their staff

Reference: WENRA Issue D 2.3

Appropriate training records and records of assessments against competence requirements shall be established and maintained for each individual with tasks important to safety.

Training programmes and facilities

Reference: WENRA Issue D 3.1

Performance based training programmes shall be established for all staff with tasks important to safety according to performed activities. The programmes shall cover basic training in order to qualify for a certain position and refresher training as needed.

Reference: WENRA Issue D 3.2

All technical staff including on-site contractors shall have a basic understanding of nuclear safety, radiation safety, fire safety, the on-site emergency arrangements and industrial safety.

Reference: WENRA Issue D 3.3

Representative simulator facilities shall be used for the training of control room operators to such an extent that the hands-on-training of normal and emergency operating procedures is effective. The simulator shall be equipped with software to cover normal operation, anticipated operational occurrences, and a range of accident conditions.

Reference: WENRA Issue D 3.4

For control room operators, initial and annual refresher training shall include training on a representative full-scope simulator. Annual refresher training shall include at least 5 days on the simulator.

Reference: WENRA Issue D 3.6

Maintenance and technical support staff including contractors shall have practical training of a requisite level on the required safety critical activities.

Training

Reference: WENRA Issue LM 6.1

Shift personnel and emergency response organizations shall be regularly trained and exercised using simulators for the emergency operation processes and, where practicable, for the severe accident management guidelines.

Reference: WENRA Issue LM 6.2

The transition from emergency operation processes to the severe accident management guidelines shall be exercised.

Reference: WENRA Issue LM 6.3

Interventions called for in severe accident management guidelines and needed to restore necessary safety functions shall be planned for and regularly exercised.

3.13.1.3 REQUIREMENTS FOR PLANNING OF ACTIVITIES IN EMERGENCY SITUATIONS

The objective is to specify the requirements for the internal emergency plan of a NPP, organization of emergency response and the links to the External Emergency Plan of NPP Temelín. In the following stage of license documentation, the section shall contain requisite changes and complementation resulting from incorporation of new units into the established organization of emergency response at the Temelín location.

Reference: Decree No. 195/1999 Coll., Article 6, SÚJB Safety Guide BN-JB-1.0 (142)

Emergency preparedness: In order to secure effectiveness of the last level of in-depth protection and requirements of special decrees (Decree No. 318/2002 Coll. [L. 26] and Government Decree No. 11/1999 Coll. [L. 283]). The project must secure corresponding technical means for controlling and implementation of intervention.

Reference: SÚJB Safety Guide BN-JB-1.0 (38)

It is necessary to assess even the course of radiation consequences of severe accidents not having the character of practically eliminated conditions:

- as a basis for drawing up manuals designed for managing of accidents and for staff training
- as a basis for executing plans for protecting the staff and population, and implementing mitigating measures to reduce the impact of radioactive releases threatening the staff, population and the environment

Reference: Safety Guide SÚJB BN-JB-1.0 (143)

The nuclear installation must be equipped with an emergency control centre (workstation for the group managing the intervention during an extraordinary event) separated from the control room and the backup equipment (supplementary control room), which must be suitably placed and protected in order to allow a sufficiently long stay and must be equipped with the means securing:

- important information about parameters of the equipment
- information on the internal and external radiological and meteorological situation
- communication with the control room or backup control equipment and other important places in the nuclear installation
- communication with the technical support centre, communication with state nuclear regulatory authorities, integrated emergency service and the respective local municipal authority and affected state authorities
- warning operators and allowing their possible sheltering

Emergency preparedness and response plan

Reference: WENRA Issue R 2.1

The plant operator shall prepare an on-site emergency plan of the NPP and establish the necessary organizational structure for clear allocation of responsibilities, authorities, and arrangements for co-ordinating plant activities and co-operating with external response agencies throughout all phases of an emergency.

Reference: WENRA Issue R 2.2

The plant operator shall provide for:

- Timely recognition and classification of emergencies
- Timely notification and alerting of response personnel
- Ensuring the safety of all persons present on the site, including the protection of the emergency workers
- Informing the authorities and the public, including timely notification and subsequent provision of information as required
- Performing assessments of the situation from the technical, and radiological points of view (on- and off-site)
- Monitoring radioactive releases

- Treatment and first aid of a limited number of contaminated and/or overexposed workers/persons on site
- Plant management and damage control

Reference: WENRA Issue R 2.3

The site emergency plan shall be based upon an assessment of reasonably foreseeable events and situations that may require protective measures on- or off-site. The plan shall also be co-ordinated with all other involved bodies and capable of extension should more improbable, severe events occur.

Reference: WENRA Issue R 5.1

Arrangements shall be made to identify the knowledge, skills, and abilities needed for personnel to perform their assigned response functions.

Reference: WENRA Issue R 5.2

Arrangements shall be made to inform all employees and all other persons present on the site of the actions to be taken in the event of an emergency.

Reference: WENRA Issue R 5.3

Training arrangements shall include basic emergency training and ongoing refresher training on an appropriate schedule and shall ensure that emergency response personnel meet the training obligations.

Reference: WENRA Issue R 5.4

The nuclear power plant site emergency plan shall be exercised at least annually. Some exercises shall be integrated to include as many as possible of the off-site organizations concerned.

Reference: WENRA Issue R 5.5

Emergency exercises shall be evaluated systematically, and the emergency preparedness arrangements and the nuclear power plant internal emergency plan shall be subject to review and updating in the light of experience gained.

3.13.1.4 IMPLEMENTATION OF PROGRAMMES FOR OPERATION STAGE

The objective is to specify requirements for creation of programmes for maintenance, tests, supervision and inspection of systems, structures and components important to safety, and the method of their designing, documenting, reviewing and implementation. In the following stage of licence documentation, the section shall contain processes of implementation of the power plant's programmes required by the regulatory body including the schedule of their implementation related to the operation stages, in which these programmes are required.

Scope and objectives

Reference: WENRA Issue K 1.1

The plant operator shall prepare and implement documented programmes of maintenance, testing, surveillance, and inspection of systems and components important to safety to ensure that their availability, reliability, and functionality remain in accordance with the design over the lifetime of the plant. These programmes shall take into account operational limits and conditions and be re-evaluated in the light of experience.

Reference: WENRA Issue K 1.2

The programmes shall include periodic inspections and tests of structures, systems and components important to safety in order to determine whether they are acceptable for continued safe operation of the plant or whether any remedial measures are necessary.

Setting up and evaluation of programmes

Reference: WENRA Issue K 2.1

The extent and frequency of preventive maintenance, testing, surveillance and inspection of structures, systems and components shall be determined through a systematic approach on the basis of

- Their importance to safety
- Their inherent reliability
- Their potential for degradation
- Operational and other relevant experience and results of condition monitoring or diagnostics

Reference: WENRA Issue K 2.2

In-service inspections of nuclear power plants shall be carried out at intervals whose length shall be chosen in order to ensure that any deterioration of the most exposed component is detected before it can lead to failure.

Reference: WENRA Issue K 2.3

Data on maintenance, testing, surveillance, and inspection of structures, systems and components shall be recorded, stored and analysed. Such records shall be reviewed to look for evidence of incipient and recurring failures, to initiate corrective maintenance and review the preventive maintenance programme accordingly.

Reference: WENRA Issue K 2.4

The maintenance programme shall be periodically reviewed in light of operating experience, and any proposed changes to the programme shall be assessed to analyse their effects on system availability, their impact on plant safety, and their conformance with applicable requirements.

Reference: WENRA Issue K 2.5

The potential impact of maintenance upon plant safety shall be assessed.

Implementation

Reference: WENRA Issue K 3.1

Structures, systems and components important to safety shall be designed to be tested, maintained, repaired and inspected or monitored periodically in terms of integrity and functional capability over the lifetime of the plant, without undue risk to workers and significant reduction in system availability. Where such provisions cannot be attained, proven alternative or indirect methods shall be specified and adequate safety precautions taken to compensate for potential undiscovered failures.

Reference: WENRA Issue K 3.2

Procedures shall be established, reviewed, and validated for all types of maintenance, testing, surveillance and inspection tasks.

Reference: WENRA Issue K 3.3

A comprehensive work planning and control system shall be implemented to ensure that maintenance, testing, surveillance and inspection work is properly authorized and carried out according to the procedures.

Reference: WENRA Issue K 3.4

Before equipment is removed from or returned to service, full consideration and approval of the proposed reconfiguration shall be ensured, followed by a documented confirmation of its correct configuration and, where appropriate, functional testing.

Reference: WENRA Issue K 3.5

The actions to be taken in response to deviations from the acceptance criteria in the maintenance, testing, surveillance and inspection tasks, shall be defined in the procedures.

Reference: WENRA Issue K 3.6

Repairs to structures, systems and components shall be devised, authorized, and carried out as promptly as practicable. Priorities shall be established with account taken first of the relative importance to safety of the defective structure, system, or component.

Reference: WENRA Issue K 3.7

Following any abnormal event due to which the safety functions and functional integrity of any component or system may have been challenged, the plant operator shall identify and revalidate the safety functions and carry out any necessary remedial actions, including inspection, testing, maintenance, and repair, as appropriate.

Reference: WENRA Issue K 3.8

The reactor coolant pressure boundary shall be subject to a system leakage test before resuming operation after a reactor outage in the course of which its leak-tightness may have been affected.

Reference: WENRA Issue K 3.9

The reactor coolant pressure boundary shall be subject to a system pressure test at or near the end of each major inspection interval.

Reference: WENRA Issue K 3.10

All items of equipment used for examinations and tests together with their accessories shall be qualified and calibrated before they are used. All equipment shall be properly identified in the calibration records, and the validity of the calibration shall be regularly verified by the licensee in accordance with the quality management system.

Reference: WENRA Issue K 3.11

Any in-service inspection process shall be qualified, in terms of required inspection area(s), method(s) of non-destructive testing, defects being sought and required effectiveness of inspections.

Reference: WENRA Issue K 3.12

When a detected flaw that exceeds the acceptance criteria is found in a sample, additional examinations shall be performed to investigate the specific problem area in the analysis of additional analogous components (or areas). The extent of further examinations shall be decided with due regard for the nature of the flaw and degree to which it affects the nuclear safety assessments for the plant or component and the potential consequences.

Reference: WENRA Issue K 3.13

Surveillance measures to verify the containment integrity shall include

- leak rate tests (pressure decrease tests)
- tests of penetration seals and closure devices such as air locks and valves that are part of the boundaries, to demonstrate their leaktightness and, where appropriate, their operability
- inspections for structural integrity (such as those performed on liner and pre-stressing tendons)

3.13.1.5 OPERATION PROCESS REQUIREMENTS

The objective is to specify the requirements regarding the scope, content, and method of drawing up of operation regulations, procedures for abnormal situations of emergency regulations and instructions for controlling of severe accidents. In the following stage of licence documentation, the section shall contain the characterisation of control documents and operation regulations used by the operating organization for safe management of normal operation and abnormal accident conditions. These are for example administrative procedures, operation and emergency procedures and maintenance and other processes including schedules of their implementation.

Objectives

Reference: WENRA Issue LM 1.1

A comprehensive set of emergency operating procedures for design basis accidents and design extension conditions, and also guidelines for severe accident management shall be provided.

Range

Reference: WENRA Issue LM 2.1

Emergency operating procedures shall be provided to cover design basis accidents and shall provide instructions for restoring the plant state to a safe condition.

Reference: WENRA Issue LM 2.2

These procedures shall also be provided to cover design extension conditions up to, but not including, the onset of core damage. The objective is to renew or compensate the lost emergency functions and launch activities which would prevent core damage.

Reference: WENRA Issue LM 2.3

Severe accident management guidelines shall be provided to mitigate the consequences of severe accidents for the cases where the measures provided by emergency operating procedures have not been successful in the prevention of core damage.

Reference: WENRA Issue LM 2.4

Emergency operating procedures for design basis accidents shall be symptom-based or a combination of symptom-based and event-based procedures. Emergency operating procedures for design extension conditions shall be only symptom-based.

Format and content of procedures and guidelines

Reference: WENRA Issue LM 3.1

Emergency operating procedures shall be developed in a systematic way and shall be supported by realistic and plant specific analysis performed for this purpose. These procedures shall be consistent with other operational procedures, such as alarm response procedures and severe accident management guidelines.

Reference: WENRA Issue LM 3.2

Emergency operating procedures shall enable the operator to recognise quickly the accident condition to which it applies. Entry and exit conditions shall be defined in the emergency operating procedures to enable operators to select the appropriate emergency operating procedure, to navigate among emergency operating procedures and to proceed from emergency operating procedures to severe accident management guidelines.

Reference: WENRA Issue LM 3.3

Severe accident management guidelines shall be developed in a systematic way using a plant specific approach. Severe accident management guidelines shall address strategies to cope with scenarios identified by the severe accident analyses.

Verification and validation

Reference: WENRA Issue LM 4.1

Emergency operating procedures and severe accident management guidelines shall be verified and validated in the form in which they shall be used in the field, so far as practicable, to ensure that they are administratively and technically correct for the plant and are compatible with the environment in which they shall be used.

Reference: WENRA Issue LM 4.2

The approach used for plant-specific validation and verification shall be documented. The effectiveness of incorporating human factors engineering principles in procedures and guidelines shall be judged when validating them. The validation of emergency operation processes shall be based on representative simulations, using a simulator, where appropriate.

Review and updating of emergency operating procedures and severe accident management guidelines

Reference: WENRA Issue LM 5.1

Operating procedures and severe accident management guidelines shall be kept updated to ensure that they remain fit for their purpose.

3.13.1.6 REQUIREMENTS FOR SECURING OF PHYSICAL PROTECTION

The objective is to specify the requirements for the method of securing of physical protection. In the following stage of licence documentation, the section shall contain the principles for creating and operating of physical protection with focus on requisite changes and complementation resulting from inclusion of new units into the set system of physical protection at the Temelín location in the course of construction and operation including the implementation schedule.

Physical protection

Reference: Decree No. 195/1999 Coll., Article 11, SÚJB Safety Guide BN-JB-1.0 (57)

The nuclear installation shall be designed in such a manner, that the physical protection of the nuclear installation and of nuclear materials may be provided in conformity with the requirements of special legal decree (Decree No. 144/1997 Coll. [L. 285]).

Control of access to the nuclear power plant

Reference: IAEA SSR 2/1 Req. 38

The nuclear power plant shall be isolated from its surroundings with a suitable layout of the various structural elements so that access to it can be controlled. Provision shall be made in the design of the buildings and the layout of the site for the control of access to the nuclear power plant by operating personnel and/or for equipment, including emergency response personnel and vehicles, with particular consideration given to guarding against the unauthorized entry of persons and goods to the plant.

Prevention of access to areas important to safety

Reference: IAEA SSR 2/1 Req. 39

Unauthorized access to, or interference with, items important to safety, including computer hardware and software, shall be prevented.

3.13.1.7 STAFF COMPETENCE

The section contains requirements for health competence of the staff. In the following stage of licence documentation, the section shall contain the reasoned scope of assurances that the system of selection and preparation guarantees that power plant staff is credible, is able to perform tasks in a reliable way and that the staff members are mentally and physically competent to perform their duties without adverse impact on safety.

Reference: WENRA Issue D 2.4

The staff members that acquired authorization for functions important to safety are subject to health checks in order to secure their competence for tasks and duties imposed on them. Health examinations are repeated at specified intervals.

3.13.2 CHARACTERISATION OF REQUIREMENTS IN THE AREA OF OPERATION FOR THE NEEDS OF PRELIMINARY ASSESSMENT

This section contains characterisation of requirements for the design in the area of securing of future operation for the needs of preliminary assessment of the design concept. The scope and depth of provided information corresponds to the stage in which the design is. The operating requirements were identified based on the technical part of the BIS stipulating the requirements for safety, technical and operational design of the future power plant. This section includes partial assessment as to whether the preliminary concept of the design segment in question complies with the requirements specified in Sections 3.13.1.1 to 3.13.1.7. The subject matter is determination and assessment of the general level of operating requirements, whereas the particular method of fulfilment of these requirements shall be specified in the nuclear power plant design and evaluated in the following stage of safety documentation.

3.13.2.1 ORGANIZATION STRUCTURE OF THE PLANT OPERATOR IN THE PRELIMINARY DESIGN CONCEPT

When delimiting competencies within the contract between the future contractor and plant operator, the area of staff organization was specified within the operator's responsibility. A requirement was stipulated for provision of recommendation for determination of optimum organization of staff by the future contractor of the power plant according to the following principles:

- the overall arrangement of the construction site and the power plant and the ensuing organizational structure shall be focused on minimization of time losses and optimisation of staff activities
- the contractor shall provide support when determining the optimum number and structure of operation and maintenance staff including shift staff
- the contractor shall provide his evaluation of requirements for operating and maintenance staff including the required qualifications
- the contractor shall provide a proposal of suitable organizational securing including requirements for personnel coverage for specific operating and maintenance operations based on experience from operation of previous analogous units

With the contractor's support, a suitable organizational structure shall be proposed and documented which shall secure safe and reliable operation of the nuclear power plant including organization and personnel securing of organization of accident response. Responsibilities, authorities, and lines of communication shall be explicitly defined and documented in the organizational structure design for all staff with duties important to safety. An established, documented, applied, maintained and evaluated system of management of ČEZ, a. s. shall be applied with the priority of continuous improvement in order to increase safety, quality and protection of the environment. Principles, methods and procedures for introduction of the management system supporting safety, which shall be applied within the design of the target organization of the power plant, are contained in Section1.7 of this Report.

Partial preliminary assessment

The preliminary concept of the design summarizing the most important requirements for operation stated in Section 3.13.2.1 creates a prerequisite for meeting of requirements contained in WENRA [L. 27] Issue B 1.1 to 1.3 in the stage of operating of the nuclear power plant.

3.13.2.2 TRAINING REQUIREMENTS IN THE PRELIMINARY DESIGN CONCEPT

The Contractor shall provide comprehensive and full preparation and training of plant operator's staff with the objective to make the staff competent for managing safe and efficient starting up, operating, inspecting, repairing and maintaining of nuclear power plant units. The plant operator shall keep within its powers the specification and selection of staff with suitable education structure and corresponding experience in order to achieve training objectives.

Preparation and training of staff shall be executed according to the programme and time schedule, which shall be approved by the plant operator.

The contractor shall be bound to prove before the start of training the conformity of its approach in the area of training and education with Czech legislation, IAEA recommendations contained in IAEA NG-T-2.2, NS-G-2.8 and TRS-380, and the proven procedures (good practice) from documents of WANO, INPO and the WENRA association.

The concept of training of workers based on a systematic approach to training (SAT)

Analyses

The contractor shall specify the number of plant operator's workers necessary for engineering activities, operating, maintenance, etc. for participation in the following stages of the project stating the number of workers for:

- designing
- construction
- starting-up
- operating

The contractor shall submit qualification requirements and job descriptions for this staff.
Technical specification of a full-scope simulator containing the definition of scope of simulation and training capabilities shall be an important part of these analyses.

Designing

The contractor shall identify requisite training for individual working positions based on the requirements for qualifications and job descriptions. The contractor shall be bound to create a list of objectives for all types of training for each group of plant operator's staff in all stages of the design.

Development

The contractor shall be bound to use SAT results in the stage of designing for creation of training programmes for individual personnel groups in each stage of the design including defining the minimum time of theoretical and practical preparation, determination of inputs and outputs, feedback models, etc. The contractor shall be bound to create further education materials according to training needs (training texts, syllabi, etc.).

Implementation

The contractor shall be bound to provide a training schedule for the plant operator's staff and shall participate in implementation of the training.
Training shall take place in relation to the particular stages of the project. Training shall be divided into 3 time stages:
1st stage (early stage) - The contractor shall secure training of the plant operator's workers specialized in activities during designing and construction and the plant operator's trainer from the education and training department.
2nd stage - In cooperation with training instructors from the training department the contractor shall secure training of the selected plant operator's start-up and operating personnel.
3rd stage - Training for other staff of the plant operator shall be provided by training instructors from the training department.

Assessment

The contractor shall be bound to assess the first two training stages in cooperation with other participants in the process to support the third stage of training of the plant operator's employees. The objective is to secure implementation of all changes resulting from assessment into the training documentation.

The contractor shall provide the plant operator with all necessary documents including all technical data and requirements necessary for corresponding education materials for training and qualification of workers before launching start-up and also necessary for requisite periodical training and renewal of qualification in the stage of future operating.

The contractor in cooperation with the plant operator shall determine the policy of preparation and training of personnel for the operating stage and shall draw up an overall continuously updated training plan based on the summary of specified needs of staff competencies and defined training objectives while respecting safety priority.

In the period of plant operation, the application of a systematic approach to training is assumed so that the process is secured starting with identification of competences required for implementation of activities of a respective job, drawing up and implementation of training programmes including processing of respective training materials and aids to achieve the required competence and then subsequent evaluation of training. Training programmes shall be drawn up for all staff implementing tasks important to safety. The programmes shall cover basic training in order to qualify for a certain position and refresher training within the requisite scope.

All operating technical employees including employees of contractors and their subcontractors active in the plant's operation and maintenance of plant equipment shall be provided with basic knowledge of nuclear, fire and general safety and organization of emergency response on the site. Maintenance and technical support staff including contractors shall be subjected to practical training of a requisite level on the required safety critical activities.

A full-scope simulator allowing systematic training of all standard and emergency processes shall be used for training of operation management staff. It shall be used for arranging of basic and annual refresher training within the minimum scope of at least 5 days of activities on the simulator. The simulator shall be equipped with software covering normal and abnormal operation and basic design basis accidents and design extension conditions.

Maintenance and technical support workers including contractors shall be subjected to practical training for required safety critical activities.

Only the qualified persons with requisite knowledge, skills and attitudes to safety shall be authorized to perform activities and acts important to safety. The licensee shall secure that the staff implementing safety related tasks including contractors shall be trained in a suitable way and shall be qualified. The licensee shall define and document requisite qualification requirements for its workers. Corresponding records of training and records of assessment (exams) for each person with safety related tasks shall be introduced and maintained.

Partial preliminary assessment

The preliminary draft of the project summarizing the most important requirements for preparation and training of staff stated in Section 3.13.2.2 creates a prerequisite for meeting of requirements of the document WENRA [L. 27] Issue D 1.1, 1.2, 2.1 to 2.3, 3.1 to 3.4, 3.6, LM 6.1 to 6.3 in the relevant designing stages, construction, starting-up and operation of the nuclear power plant.

3.13.2.3 REQUIREMENTS FOR PLANNING OF ACTIVITIES IN EMERGENCY SITUATIONS IN THE PRELIMINARY DESIGN CONCEPT

Within the implementation of the design, corresponding means shall be secured to support managing of abnormal conditions and accident conditions including severe accidents. Within the power plant design, the elements such as technical support centre, gathering place, emergency routes and exits as well as measures for emergency communication and areas for emergency equipment and facilities shall be considered.

If it is impossible to prevent heavy damage to the fuel system, the design must provide a sufficient time for obtaining of requisite expert opinions for the site personnel responsible for managing of the emergency condition and for organizing of measures within the emergency plans. The design shall meet the minimum requirements for power plant autonomy for external supplies (heat extraction, power supply).

The licensee shall secure within the deadline according to the schedule of construction of respective units extending of the existing internal emergency plan of ETE 1,2, together with complementation of organization structure while preserving a clear definition of responsibilities, authorization and measures for coordination of activities on the power plant and cooperation with external organizations in the course of an accident. As part of that, measures for application of a basic training system, follow-up regular training, and practicing of the Internal Emergency Plan of the Nuclear Plant shall be adopted together with systematic evaluation and updating in conformity with acquired experience.

The Common Internal Emergency Plan of the Nuclear Power Plant shall be based on the assessment of anticipated events and situations of all units operated on site which may require protection measures on the NPP Temelín site area. The plan shall also be coordinated with all the other concerned authorities and shall even cover the proceedings in the conditions of a less likely serious event (severe accident).

On site the licensee shall secure for all operated units the following:

- timely identification and classification of accidents
- timely notification and warning to responsible personnel
- ensuring the safety of all persons present on site, including protection of emergency workers
- notification of authorities and organizations, warning to the general population including timely notification and providing of follow-up information as necessary
- evaluation of the situation in terms of the technical and radiological point of view (on- and off-site)
- Monitoring radioactive releases
- providing treatment and first aid to contaminated or exposure workers/persons
- management of emergency conditions of the power plant and urgent remedial measures

In order to provide support and to implement requisite interventions when dealing with emergency conditions, a sufficient scope of equipment shall be provided (including equipment allowing monitoring of response), whose operability shall be demonstrated in the conditions corresponding to its application.

The power plant arrangement shall allow access in the post-emergency condition in the scope necessary for supervision and restoration operations. The plant contractor shall draw up a plan that identifies the areas to which access is required and shall secure this requirement by technical means and by proposing organizational measures.

Partial preliminary assessment

The preliminary concept of the project summarizing the most important requirements for operation stated in Section 3.13.2.3 creates a prerequisite for meeting the requirements of the document of Decree No. 195/1999 Coll. [L. 266], Article 6; Safety Guide SÚJB BN-JB-1.0 [L. 276] (38, 142, 143); WENRA [L. 27] Issue R 2.1 - 2.3, 5.1 – 5.5. in the stage of operation of the nuclear power plant.

3.13.2.4 IMPLEMENTATION OF PROGRAMMES FOR THE OPERATION STAGE IN THE PRELIMINARY DESIGN CONCEPT

The contractor shall provide a summary of personnel and administrative policies and procedures drafted in order that the organization of operation and maintenance of the power plant is sufficiently sized and structured and efficient management is possible together with fulfilment of all activities concerning operation and maintenance including the possibility to assess the effect of maintenance on safety.

The contractor shall systematically specify processes of installation maintenance including defining of requirements for expertise, tools, test equipment, access possibilities, environmental conditions, etc. These tasks shall also include all tests necessary for returning of equipment into operation after termination of maintenance. The project shall allow introduction of a planning and control system to ensure that maintenance, testing, surveillance and inspection work is properly authorized and carried out according to the specifications.

The provided documents shall allow the plant operator in cooperation with the contractor to prepare and activate the documented programmes for maintenance, tests, surveillance and inspection of systems, structures and components important to safety so that their preparedness, reliability and operability is in conformity with the design over the lifetime of the power plant. These programmes shall take into account operational limits and conditions and shall be reviewed based on experience whereas all the proposed changes shall be analysed with regard to their impact on the operability of the system, power plant safety and their conformity with effective legislation requirements.

Assessment of requirements for the scope of maintenance which shall be dealt with by both the contractor and the plant operator shall be implemented in the course of designing so that corresponding balance is achieved between the designing requirements and maintenance requirements. The result of this assessment shall be documented. The design shall be drafted so that the scope and complexity of maintenance activities over the lifetime of the power plant is simplified and reduced. The process of repairs of structures, systems and components shall be optimised with the objective to minimize the period for planning, approving and implementation of repairs. Priorities of remedial activities shall be established with account taken first of the relative importance to safety of the defective structures, systems, or components.

The power plant design shall be drafted with the objective to optimise maintainability by including the human factor and by securing corresponding premises, lighting, platforms, lifting devices, handling equipment, screening, communications, HVAC and servicing. Maintenance activities shall be optimised in terms of minimization of radiation dosages. The actions to be taken in response to deviations from the acceptance criteria in the maintenance, testing, surveillance and inspection tasks, shall be defined.

Measures shall be taken to facilitate critical maintenance activities. The contractor shall determine and propose tools for training and exercise such as mock-ups and CAD for critical equipment in order to facilitate training, minimize the possibility of error occurrence during maintenance and decrease staff exposure to increased radiation. The project shall specify and state specific support systems requisite for maintenance of critical activities taking into account all aspects of maintenance work and special handling equipment. At places with increased radiation, the use of robotic systems shall be considered.

Attention shall be paid to the facilitation of maintenance of equipment on site and easy replaceability of components. Similarly, at places where modular structures are used, attention shall be paid to the maintenance of all components on site or replacement of the whole module.

Accessibility of equipment (orientation and surroundings) shall be at such a level that it facilitates maintenance operations particularly of critical components or those that are regularly replaced.

Procedures shall be established, reviewed, and validated for all types of maintenance, testing, surveillance and inspection tasks.

The contractor shall stipulate a preventive maintenance programme (based on operational experience) and predictive maintenance (based on anticipated behaviour), with the objective to secure optimum operation and safety of power plant equipment. Systems and their components shall be selected with regard to the designed lifetime of the power plant. These elements shall allow the operator to implement maintenance programmes based on reliability centred maintenance (RCM). Preventive maintenance programmes shall be based on a suitable scope of checks and maintenance codes. Predictive maintenance shall be based on monitoring of equipment status. The selection of representative equipment for these programmes shall be based on assessment according to PSA. The scope and frequency of preventive maintenance, tests, checks and inspections of systems, structures, and components shall be determined based on a systematic approach and based on their importance to safety, reliability according to diagnostics outputs, potential degradation and operational and other experience. The provided system of programmes shall allow recording, storing and analysing of the data from maintenance, tests, checks and inspections of structures, systems, and components. These records shall be assessed with the objective to find (ascertain) proofs of incipient and recurring failures and according to that initiate remedial maintenance and also with the objective to reassess preventive maintenance programmes.

Already from drawing up the design bases, the project shall be designed with the objective to achieve the design lifetime of the plant. This shall mean considering the mechanisms of ageing and wearing of components in the project. Checks and inspections shall be established together with the required scope of maintenance including replacement of structures, systems, and components within the scope supporting this target. In the case that it is impossible to maintain integrity and operability of structures, systems, and components, proven substitute or indirect methods shall be specified and adequate safety measures shall be adopted to compensate possible undetected fails.

The power plant shall be equipped with corresponding support equipment for the activities of workers and maintenance of equipment. This section deals with the requirements for access to radioactive workplaces without stating the requirements for power plant servicing, workshops for contaminated parts and a clean workshop (i.e. radioactive and non-radioactive), and management of spare parts.

Based on the experience with maintenance of the operated power plants and production specifications, the contractor shall specify a list of recommended spare parts for the whole power plant. It shall secure the primary storing and detailed specifications for purchasing of refurbished parts. The list shall be approved by the plant operator with the focus on spare parts of primary equipment, safety qualification, storability, and requirements for maintainability of equipment over the lifetime of the plant.

The power plant design shall secure means for facilitation of replacement of all main components with the exception of the reactor pressure vessel and basic structures of the power plant in conformity with the requirements for the designed plant lifetime.

The scope of standard operational checks shall be specified within the design. A policy shall be defined and a programme provided for periodical checks defining the frequency of tests and methods for shortening or extending of testing intervals. The programme shall include periodical checks and tests of systems, structures and components important to safety with the objective to prove their reliability and to determine whether they are acceptable for continuing of safe operation of the power plant or remedial measures are necessary.

The power plant project shall allow to a maximum possible extent that the periodical operational tests be performed during power operation without disrupting standard operation, with the exception of cases when this is counterproductive. These tests shall be designed so that their implementation does not impair safety functions. It shall be possible to test systems and subsystems or instrumentation loops in the conditions similar to operating conditions as much as possible. Operational checks of the power plant shall be performed at a frequency that guarantees that damage to the most loaded component is ascertained before its failure.

The check techniques used in the course of production shall include techniques compatible with techniques used in operational checks in the operation stage.

The methods of technical checks applied in the operation stage shall be based on methodologies of checks applied in the stage of production of structures, systems, components, which shall create a reference basis for operating checks. The transfer of requisite know-how shall be secured in the form of additional training of the plant operator's staff.

All materials shall be manufactured in conformity with material specifications for individual components. These specifications shall define checks and tests in conformity with the approved industrial standards.

With all materials being a part of components of safety category 1 whose disruption may have an impact on operating staff safety or the environment, mechanical properties and chemical composition shall be established to secure conformity with material specifications.

Materials of components strained by internal overpressure including integrated supports and welds shall be tested by non-destructive methods according to respective standards, in particular by ultrasound, radiographic, magnetic and liquid penetrant techniques.

Non-destructive test methods (NDT) and the processes used for checks of material in the course of production shall be harmonized with the methods and processes of operational checks.

In the case of important components, it is recommended that operational non-destructive tests should include a pre-operation check (basic or initial level). This basic level shall be derived with regard to the results from checks obtained in the course of production.

The reactor pressure system shall be subjected to tightness tests before restarting after shutdown (during which time tightness might be affected) and periodically in conformity with specified inspection interval.

Software, special procedures including checking procedures and the workers utilized for or performing functions associated with safety related activities shall be subject to a corresponding process of validation or qualification. All items of equipment used for examinations and tests together with their accessories shall be qualified and calibrated before they are used. All equipment shall be properly identified in the calibration records, and the validity of the calibration shall be regularly verified by the licensee in accordance with the quality management system. Any in-service inspection process shall be qualified, in terms of required inspection area(s), method(s) of non-destructive testing, defects being sought and required effectiveness of inspections.

Conditions shall be created for pre-operation and periodical integral tests of containment leaktightness. They shall be performed at the full test pressure.

The design shall include measures that shall allow at each refuelling local individual tests of:

- each passing of piping through the primary containment in the section between the dividing containment
- mechanical and electrical penetrations
- substantial openings such as assembly openings or hermetic air locks, etc.

The contractor shall provide suitable instrumentation for securing of the pre-operation and operation programme of surveillance, which shall allow checking and evaluation of parameters such as thermal impact of the structure, deformation of a structure, deformation tension, and prestress level.

Specifications for equipment purchase shall provide detailed check processes which shall be used during production and on site or they shall be provided by the equipment supplier.

PSA and methods of risk assessment shall be used as a tool for optimisation of the check and inspection programme.

The frequency and time demands of periodical checks shall be optimised without negative impact on safety.

Partial preliminary assessment

The preliminary concept of the project summarizing the most important requirements for operation stated in section 3.13.2.4 create a prerequisite for meeting the requirements of the document WENRA [L. 27] Issue K 1.1, 1.2, 2.1 to 2.5, 3.1 to 3.13 in the stage of operation of the nuclear power plant.

3.13.2.5 REQUIREMENTS FOR OPERATION PROCEDURES IN THE PRELIMINARY DESIGN CONCEPT

The designer shall hand over for approval operation and maintenance procedures and shall provide instructions and means for their processing and maintaining. The procedures shall include technical specifications, drawings, rules and principles for operation, tests, and maintenance, etc. These specifications shall take into account the human factor as an integral part of man-machine interface in order to decrease the potential for operator's error and to improve operation and maintenance activities.

The project shall contain verification securing that procedures and instructions cover all required functions and tasks. Verification and validation of procedures and instructions shall be implemented so as to secure that they are correct for the power plant in terms of technical and administrative aspects and that they are in conformity with the conditions in which they are applied. The effectiveness of incorporating human factor engineering principles in procedures and guidelines must be judged when validating them. The validation of emergency operating procedures shall be based on representative simulations, where appropriate. At control room service stations and also at other suitable workplaces using computer means, the procedures should be available in electronic form on displays. These computerized processes shall support operators when performing and/or verifying tasks.

Besides the electronic version, printed versions of procedures shall be used at selected workplaces. The procedures shall be clear, unambiguous and easy to use. The method of processing of the specifications and arrangement of individual steps must not allow erroneous interpretation.

Operating procedures and severe accident management guidelines shall be kept updated to ensure that they remain fit for their purpose.

The following types of operation procedures shall be drawn up for standard operation modes:

- procedures for operation of systems (e.g. start-up of systems, periodical tests)
- procedures for power plant operation (for example power plant start-up, additional cooling, etc.)
- procedures for indication response (response to alarms)
- equipment maintenance procedures

For abnormal operation modes such as sudden loss of output, equipment failures, abnormal operation modes shall be drawn up. A combination of symptom based and event based approach is assumed for this type of operation procedures.

Procedures for design basis accidents shall be processed systematically and shall be based on analyses drawn up for this purpose. These procedures shall be in conformity with other operating procedures, particularly those that concern reaction to warning signals (response to alarms) and instructions for management of serious accidents. Emergency procedures shall allow operators to quickly recognize accident conditions for which they are intended. These procedures shall specify input and output conditions so that they allow the operators to select the suitable procedure, find their way in the procedures, determine transition between procedures for design basis accidents, or transition to instructions for management of serious accidents.

A symptom oriented approach in emergency procedures shall be based on the fact that the corresponding solution is selected based on the actual development of the emergency condition, which is identified based on unambiguous symptoms characterizing the given event or a group of events. If symptoms change during solving of an emergency condition and the applied strategy may be no longer used, then the structure of these procedures shall allow leaving this strategy and continuing in solving the emergency condition using another strategy (procedure) which better corresponds to the newly occurred conditions. Continuous diagnostics of unit condition in the course of dealing with an emergency situation thus allows the operating personnel to react correctly to the possible changes of input conditions in the respective procedure according to the development of the emergency situation and their interventions are thus always the optimum reaction to the given condition of the unit.

The contractor shall submit to the plant operator for approval an accident management system, which shall contain preplanned, for this purpose drawn up operating procedures and instructions (engineering decisions) which in design basis and design extension conditions (with the maximum use of existing structures, systems, and components) allow renewal of checking of technology status. A comprehensive set of emergency procedures and instructions for management of severe accidents shall guide the operating staff in design basis and design extension conditions for securing of fundamental safety function and mitigating of consequences of serious accidents. The accident management system shall be based on the IAEA NS-G-2.15 Guide.

The objective of emergency operating procedures is to renew or replace the loss of safety functions and to determine the activities leading to minimization of accident consequences in all project considered operating modes. Emergency operating procedures shall be drawn up as symptom oriented, providing instructions to the operative controlling staff for dealing with emergency situations with the objective to renew the safe status. Emergency operating procedures shall also contain instructions for dealing with combined and multiple fail sequences. Further, instructions shall be drawn up for management of severe accidents as a help for operators when dealing with these conditions. Severe accident management instructions shall be drawn up as symptom oriented. The instructions shall contain a strategy for the management of scenarios, determined by analyses of serious accidents.

These instructions shall cover all design considered operating modes (e.g. power operation, hot shutdown mode and cold shutdown mode) and both external and internal risks (e.g. earthquake, plane crash).

Drawing up of instructions for management of severe accidents shall be required based on:

- in-depth analysis of the human factor,
- links to procedures or renewal of safe status and renewal of safety critical functions
- links to emergency plans
- analyses of serious accident scenarios

Serious accident management instructions shall be draw up with the objective to secure fulfilment of the following objectives:

- termination of release of fission products from the power plant
- maintenance (or returning) of containment to a controlled, stable condition
- returning of the active zone to a controlled, stable condition
- minimization of release of fission products
- maximization of renewal of operability of the equipment and monitoring means

Unambiguous criteria shall be attached to the entry conditions from the above specified emergency operating procedures of instructions for management of severe accidents and the time interval of all interventions shall be defined.

Partial preliminary assessment

The preliminary concept of the project summarizing the most important requirements for operation stated in Section 3.13.2.5 creates a prerequisite for meeting the requirements of the document WENRA [L. 27] Issue LM 1.1, 2.1 to 2.4, 3.1 to 3.3, 4.1, 4.2 and 5.1 in the stage of operation of the nuclear power plant.

3.13.2.6 REQUIREMENTS FOR SECURING OF PHYSICAL PROTECTION IN THE PRELIMINARY DESIGN CONCEPT

The design of physical protection shall be designed in order to provide the required way of physical protection of the nuclear installation and nuclear material.

Physical protection of the power plant shall fully respect the Analysis of possibilities and needs of physical protection drawn up by course of provisions of Article 13 (3)(d) of Act No. 18/1997 Coll. [L. 2] as a part of application for licence and the content of this separate documentation shall correspond to the requirements of appendix A. II. of this Act.

Partial preliminary assessment

The preliminary concept of the project summarizing the most important requirements for operation stated in section 3.13.2.6 creates a prerequisite for meeting the requirements of documents 195/1999 Coll. [L. 266], Article 11, Safety Guide SÚJB BN-JB-1.0 [L. 276] (144); IAEA SSR 2/1 [L. 252] Req. 38 and 39 and WENRA [L. 27] Issue D 2.4 in the stage of operating of the nuclear power plant (in detail see Analysis of needs and possibility of securing of physical protection).

3.13.2.7 STAFF COMPETENCE IN THE PRELIMINARY DESIGN CONCEPT OF THE PROJECT

The staff that acquires authorization for functions important in terms of safety shall be subjected to health checks in order to secure their competence for tasks and duties imposed on them. Health examinations shall be repeated at specified intervals.

Partial preliminary assessment

The preliminary concept of the project summarizing the most important requirements for operation stated in section 3.13.2.7 creates a prerequisite for meeting the requirements contained in WENRA [L. 27] Issue D 2.4 in the stage of operating of the nuclear power plant.

3.13.3 SUMMARY PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT IN THE AREA OF OPERATION REQUIREMENTS

The ETE 3,4 design shall meet the safety requirements specified in Section 3.3.1.1 as well as the requirements specified in Sections 3.13.1.1 to 3.13.1.7, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and related implementing decrees as regards nuclear safety, radiation protection and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270] documents.

The principles of the design solution for the reactor facility described in Section 3.13.2 , were set up based on the requirements placed by the applicant for licence on the potential suppliers of the nuclear facility within the tender, and make up the concept of the design solution for this part of the design. Implemented partial assessments confirm that the selected project solution in the area of operation characteristics of the future nuclear source create prerequisites for meeting specific requirements for operation of systems, structures, components and nuclear unit as a whole specified by Decree No. 195/1999 Coll. [L. 266], Safety Guide SÚJB BN-JB-1.0 [L. 276], document IAEA SSR 2/1 [L. 252] and document WENRA [L. 27]. The preliminary concept of operation suits the implemented preliminary evaluation of design concept.

3.14 THE PARTICULAR METHOD OF IMPLEMENTATION OF INDIVIDUAL REQUIREMENTS FOR OPERATION STATED IN SECTION 3.13.2 SHALL BE DRAWN UP IN DETAIL IN THE POSR DRAWN UP BASED ON THE SELECTED DESIGN.TEST PROGRAMME DURING START-UP

The specified legislation does not contain the requirements for test programmes during start-up. This shall be specified within the following stages of the licensing documentation.

3.15 ANALYSES OF TRANSITION EVENTS AND DESIGN BASIS ACCIDENTS

This comprehensive part of the ISAR focuses on deterministic analyses of events of abnormal operation and design basis accidents. Probability analyses and deterministic analyses of design extension conditions including severe accidents are included in Section 3.19.

Section 3.15 is divided into three basic parts.

The introduction Section 3.15.1 summarizes, analyses and specifies basic legislation requirements for Analyses of transient conditions and design basis accidents with the focus on general objectives, classification of transient conditions and accidents, characteristics of the power plant considered in safety analyses, interventions of operating staff, evaluation of individual initiation occurrences and analyses of fire risk. These requirements are defined in conformity with the existing SÚJB acts, decrees and guides complemented by relevant IAEA safety standards and WENRA reference levels.

The following Section 3.15.2 contains a description of the method of fulfilment of basic requirements based from the introduction Section 3.15.1 in a form which creates the envelope of applied requirements for implementation of Analysis of Transient Conditions and design basis accidents within all designs that may be considered within the current tender. The objective of the section is to formulate general characteristics of the design for the purposes of the partial preliminary procedure.

The final Section 3.15.3 contains a comprehensive preliminary evaluation of the design concept in light of Analysis of transient conditions and design basis accidents, summarising the conclusions of the partial preliminary assessments completed in Section 3.15.2. Within the assessment of the sum of requirements so obtained, this section contains a preliminary evaluation of the design concept as required by the law. The actual analyses of the said group of events drawn up in conformity with the principles stated further, shall be a part of the PSAR to be submitted to SÚJB together with the requirement for permission to build a new nuclear source.

3.15.1 BASIC LEGISLATION REQUIREMENTS FOR ANALYSES OF TRANSIENT CONDITIONS AND DESIGN BASIS ACCIDENTS

Reference: Safety Guide SÚJB BN-JB-1.0 (28), IAEA GS-R-4 Req. 21, 4.64, 4.66

The analysis shall contain:

- a justification for the selection of the anticipated operational events and accidents considered in the analysis
- an overview and necessary details of the collection of data, the way of modelling, the computer codes and the assumptions made
- acceptance criteria used for the evaluation of the modelling results
- results of the analysis covering the performance of the facility or activity, the radiation risks incurred and a discussion of the underlying uncertainties

- conclusions on the acceptability of the level of safety achieved and the identification of necessary improvements and additional measures

Safety analyses shall be performed in conformity with the requirements of the State Office for Nuclear Safety and documented in a sufficient scope and detail so that sufficient information is available for independent verification of safety analyses on the applicant's side and for checking and verification by supervisory organization and for updating of analyses with regard to operational experience, new information, current status of science and technology and evaluation methods.

Reference: Safety Guide SÚJB BN-JB-1.0 (27), IAEA GS-R-4 Req. 15

Safety of the nuclear power plant design shall be demonstrated by deterministic and probabilistic safety analyses. The ability of the facility to perform within the scope of project assumptions, the requirements for nuclear safety and radiation protection during normal and abnormal operation and in design basis conditions must be verified by deterministic methods. The probabilistic methods must be used to assess the risk related to the design solution as well as the operation of the nuclear installation.

Reference: IAEA GS-R-4 4.63

The quantitative and qualitative outcomes of the safety assessment form the basis for the safety report. The outcomes of the safety assessment are supplemented by supporting evidence for and reasoning about the robustness and reliability of the safety assessment and its assumptions, (including information on the performance of individual components of systems) and securing of sufficient safety margins.

3.15.1.1 GENERAL PRINCIPLES AND OBJECTIVES OF ASSESSMENTS

Reference: IAEA GS-R-4 4.62

The objective of safety assessments is to check whether the nuclear power plant is to be implemented in conformity with relevant safety requirements. The used methods, results and conclusions of safety assessments (analyses) shall be documented in such a scope which corresponds to the comprehensive character of a nuclear power plant and the ensuing radiation risk. The assessments must prove that the power plant is designed particularly in conformity with acts and decrees of the Czech Republic namely Decree No. 195/1999 Coll. [L. 266], but also in conformity with basic safety principles and relevant safety requirements IAEA and WENRA reference levels.

Reference: Safety Guide SÚJB BN-JB-1.0 (29), IAEA GS-R-4 4.48

The safety assessments must also prove that there are adequate safety margins in the design and operation of the facility securing that there is a wide margin to failure of any structures, systems and components for any of the anticipated operational occurrences or any possible accident conditions. Safety assessment results must prove that acceptance criteria for each aspect of the safety analysis are such that an adequate safety margin is ensured.

Reference: Safety Instruction SÚJB BN-JB-1.0 (29), 2/1 5.75 (1) - (5)

The deterministic safety analysis of design modes of the power plant shall mainly provide:

- establishment and confirmation of the design bases of the nuclear facility as a whole and of its individual parts
- characterization of the postulated initiating events that are appropriate for the site and the design of the plant
- analysis and evaluation of event sequences that result from postulated initiating events, to confirm the qualification requirements
- demonstration that the postulated initiating events including their development are manageable by automatic actuation of safety systems in combination with prescribed actions by the operator
- demonstration of correctness of setting of control and protection systems; acceptability of their response, including taking into account operating staff interventions
- demonstration that possible disruptive effects of systems not related to safety or false inclusion of protection systems or operator errors have been duly taken into account

Reference: IAEA SSR 2/1 5.8

The safety analysis must verify that the expected behaviour of the plant in any postulated initiating event shall be such that the following conditions can be achieved, in order of priority:

- a postulated initiating event would produce no safety significant effects or would produce only a change towards safe plant conditions by means of inherent characteristics of the plant
- following a postulated initiating event, the plant would be rendered safe by means of passive safety features or by the action of systems that are operating continuously in the state necessary to control the postulated initiating event
- following a postulated initiating event, the plant would be rendered safe by the actuation of safety systems that need to be brought into operation in response to the postulated initiating event
- following a postulated initiating event, the plant would be rendered safe by specified actions of operators in line with specifications

Reference: Decree No. 195/1999 Coll., Article 4 (3), IAEA GS-R-4 4.60, IAEA SSR 2/1 5.74

Any computing programs, analytical methods, calculational assumptions and power plant models shall be verified and validated. The users of computer codes shall have sufficient experience with the use of computing programs for the given type of facility.

3.15.1.2 CLASSIFICATION OF TRANSIENT EVENTS AND ACCIDENTS

Reference: WENRA App. E 5.1, 8.6, IAEA GS-R-4 4.51, 4.52

Safety analyses shall be based on postulated initiation events, i.e. possible deviations from normal operating conditions the development of which may result in abnormal operation or emergency conditions that may have consequences for physical barriers for confining the radioactive material or that otherwise give rise to radiation risks. The features, events and processes are to be selected on the basis of a systematic approach, and justification has to be provided that the list of occurrences is sufficiently comprehensive. This process utilizes relevant operational experience from similar facilities in terms of the reason for initiating events, their course and effects, their seriousness and efficiency of proposed remedial measures. The actual postulated initiating event is not an accident yet, however, depending on other failures it may develop into abnormal operation event, design basis accident or severe accident. Any defect formed due to a postulated initiating event is regarded as a part of this event. A generally postulated initiating event may be formed due a failure of equipment (partial or complete), staff error, internal impacts and external impacts (hazards) either of natural origin or caused by human failure.

Reference: Safety Guide SÚJB BN-JB-1.0 (23), (46), WENRA App. E 7.1

Using deterministic and probabilistic methods, engineering judgement or their combination a list of postulated initiating events shall be made that may significantly affect safety of the nuclear installation including those that may be caused by internal or external impacts induced by natural effects or human activity or by a combination of these. This list shall also include combinations of initiating events whose seriousness and probability of appearance correspond to the seriousness and frequency of appearance of design occurrences. This list must include at least the following events:

Simple postulated initiating events characteristic for nuclear facilities with water pressure reactors

- bursting of primary circuit piping with small, medium and large diameter (including the largest diameter in the reactor cooling system)
- bursting of main steam or feed piping lines
- forced reduction of flow rate of coolant through the reactor
- decreasing or increasing of flow rate of feed water
- decreasing or increasing of steam flow rate from steam generators
- erroneous opening of valves of volume compensators
- inadvertent activation of emergency cooling of the active zone
- erroneous opening of steam generator valves
- erroneous closing of valves on steam lines
- bursting of pipes or other disruption of the water pressure part of steam generator
- uncontrolled movement of controlling elements of reactor
- uncontrolled pulling or shooting of control bodies from the active zone

- malfunction of reactor replenishment system and chemical regime control
- pipe bursting, leaking or opening of a valve/fitting or leaking of thermal exchanger in the system connected to the primary circuit and placed partially outside the containment
- occurrences linked to fuel handling
- loss of external electric power supply
- falls of loads caused by malfunction of lifting devices

Postulated initiating events occurrences which may follow another initiating event

- fire
- explosion
- flooding

Subsequent events which may be induced by the postulated initiating event

- internal missiles including turbine debris
- leaking liquid (oil, etc.) from damaged systems
- vibrations
- piping whip
- effects of released medium current

Reference: Safety Guide SÚJB BN-JB-1.0 (22), IAEA SSR 2/1 5.9, IAEA GS-R-4 4.51

The postulated initiation occurrences shall be divided into a final number of representative categories (power plant conditions) formed by the most safety limiting occurrences and processes based on which design limits for systems and components important to safety shall be specified.

Reference: Decree No. 195/1999 Coll., Article 10 (1), Safety Guide SÚJB BN-JB-1.0 (24), IAEA SSR 2/1 5.16, 5.17

Analyses must demonstrate that the equipment important to nuclear safety of the nuclear installation is designed so that in the case of natural events including their combination which may not be practically excluded (earthquake, storm, floods, extreme external temperatures, extreme temperatures of cooling water, meteorological precipitation of all forms, humidity, icing, flora and fauna effects, etc.), or events caused by human activities outside the nuclear installation that may not be practically excluded (plane crash, explosions, fires, traffic and industrial accidents in the surroundings of the nuclear installation potentially resulting in fires, explosions and other hazards, electromagnetic interference or other influence of technical equipment existing outside the nuclear installation, etc.) have not been threatened - particularly fundamental safety function.

Reference: Decree No. 195/1999 Coll., Article 10 (2); SÚJB Safety Guide BN-JB-1.0 (62) WENRA App. E 6.1, IAEA SSR 2/1 5.32

When performing safety analysis it is necessary to consider:

- characteristics of the site on which the nuclear facility is to be placed
- the most significant natural events or occurrences caused by human activity, historically recorded on the site and its surroundings extrapolated taking into account limited accuracy of values and time
- combination of effects of natural events or occurrences caused by human activities and the states of abnormal operation or accident conditions caused by these phenomena or occurrences

If the results of engineering judgements, deterministic or probabilistic considerations result in the conclusion that the combination of occurrences might result in events of abnormal operation or accident conditions, then such combination of events shall be analysed as a design basis accident or shall be analysed as design extension conditions depending particularly on the likelihood of its appearance. Some events may be the result of other occurrences such as floods or earthquake. Such subsequent effects shall be regarded as a part of the original postulated initiating event.

3.15.1.3 POWER PLANT CHARACTERISTICS CONSIDERED IN SAFETY ANALYSES

Reference: Safety Guide SÚJB BN-JB-1.0 (29), IAEA SSR 2/1 5.26, IAEA GS-R-4 4.54

Deterministic safety analyses must prove observance of all acceptance criteria with sufficient safety margins. A conservative approach (application of a set of conservative deterministic rules and requirements) in safety analyses is a possible way of compensating the uncertainty in the output of the plant and staff activities. Besides conservative selection of acceptability criteria, the conservative approach includes the use of conservative assumptions, models and entry parameters for analyses including assumptions of certain failures of safety systems.

Reference: Safety Guide SÚJB BN-JB-1.0 (35), WENRA App. E 8.1, E 8.7

For safety analyses of abnormal operation and design basis accidents based on simple initiating events, it is necessary to use the least favourable (conservative) initial and limit conditions. In specific cases, it is necessary to evaluate the impact of uncertainty of input data and consequences of uncertainties of analysis methods must be evaluated, which are important to the result in the analysed cases. Proven methodology needs to be used for evaluation of uncertainty of data and models.

Reference: IAEA GS-R-4 Req. 17, 4.58, 4.59

There shall always be uncertainties associated with parameter predictions that shall depend on the nature of the facility or activity and the complexity of the safety analysis. Uncertainties in the safety analysis have to be characterized with respect to their source, nature and degree, using quantitative methods, professional judgement or both. Uncertainties that may have implications for the outcome of the safety analysis and for decisions made on that basis are to be subject to uncertainty and sensitivity analyses. Uncertainty analysis refers mainly to the statistical combination and propagation of uncertainties in data, whereas sensitivity analysis refers to the sensitivity of results to major assumptions about parameters, scenarios or modelling.

Reference: Safety Guide SÚJB BN-JB-1.0 (32), WENRA App. E 8.3

In the case of safety analyses of abnormal operation and design basis accidents, the assumption is that in order to manage the analysed designed occurrence, i.e. to transfer the reactor into a stabilized condition, only safety systems may be used in safety classes in conformity with the special procedure Decree No. 132/2008 Coll. [L. 258]) and with guaranteed reliability. When managing an abnormal operation event and a design basis accident, the functions of other active systems are to be considered only when they impair the course of events. This means that in the course of managing an abnormal operation event and a design basis accident the functions of active systems not classified as selected equipment in conformity with requirements of a special procedure are not considered or their effect is considered before the formation and course of the occurrence in a way which is least favourable for managing of the occurrence.

Reference: Safety Guide SÚJB BN-JB-1.0 (33), (34, WENRA App. E 8.2, 8.4, 8.5, 8.6

Other assumptions for safety analyses of abnormal operation events and design basis accidents:

- Respecting of the impact of safety systems and their characteristics so that their operation results in the least favourable case of course of events
- Any failure due to a postulated initiating event is a part of the original initiation occurrence
- The appearance of the most severe simple breakdown of any system component performing the requisite safety function and all other failures caused by it or failures induced by the postulated initiating event either at the moment of formation of the event or later at the least suitable moment and at the least suitable configuration of equipment implementing safety functions. It is not necessary to consider breakdowns of passive components if their occurrence may be regarded as practically eliminated excluded or their function is not concerned by the postulated event
- Jamming of one control component of the active zone (as additional impairing failure) with the highest efficiency in a hot power-free condition in order to establish a sufficient safety margin for reactor shutdown. Conservativeness of the analysis of introduction of (negative) reactivity during rapid shutdown of a reactor shall be further increased by using a conservative time delay of dropping of control components and conservative dependence of the introduced reactivity on rod position
- The loss of operation and reserve sources of power supply at the moment with least favourable impact on the course and consequences of the accident shall be considered as another conservative failure, unless the loss of power supply is considered as a subsequent failure (e.g. due to disintegration of the external grid).

Reference: IAEA GS-R-4 4.59

If the method of adequate selection of conservative parameters is not obvious in relation to the given type of scenario or based on other available analyses, this selection shall be supported by relevant sensitivity analyses. With regard to the possible impacts of uncertainties on the conclusions of safety analyses and with regard to the fact that this is a new type of nuclear power plant for the CR, in addition to the standard conservative safety analysis of events of abnormal operation and design basis accidents quantification of uncertainties shall be implemented at least for those selected design basis accidents from each group of events for which the difference between the conservative prediction and the acceptance criteria is the smallest. In these cases it shall be required that both approaches confirm observance of the relevant acceptance criteria. If the quantification of uncertainties for selected design basis accidents is not a part of initial safety analyses, then such quantification shall be implemented within independent verification of analyses on the plant operator's side.

3.15.1.4 OPERATING STAFF INTERVENTIONS

Reference: IAEA GS-R-4 Req. 11, 4.38, 4.40, IAEA SSR 2/1 5.13

Besides automatic safety systems, the safety of a nuclear power plant depends on operating staff interventions. It is necessary to consider and evaluate in safety analyses all interventions of operators that are necessary for timely diagnosing of the power plant state and for putting it into a stable and shutdown mode using instrumentation for monitoring of the status and for monitoring of manual interventions. Safety analyses must prove that the requirements related to the human factor while considering human-machine interactions have been duly taken into account in the power plant design. It is necessary to confirm that the safety measures and operational measures for standard operation and for abnormal operation events and for accident conditions secure a corresponding safety level.

Reference: IAEA SSR 2/1 5.12, WENRA App. E 9.3

Where prompt and reliable action is necessary, with the objective of preventing development of an event into more serious conditions endangering integrity of subsequent barriers, the initiation of this intervention by safety systems must be automatic. The possibility of manual starting of systems and other operating staff interventions shall be considered only provided that there is sufficient time for making a decision and implementing the intervention and that in order to implement this intervention, reliably adequate and clearly defined procedures (administrative, operational, emergency) are available, which are regularly practised. An assessment shall be made of the potential for an operator to worsen an event sequence through erroneous operation of equipment or incorrect diagnosis of the necessary recovery process.

Reference: SÚJB Safety Guide BN-JB-1.0 (46)

The safety analysis shall check that if the implementation of functions of safety systems is not activated by protection systems and automatically controlled or implemented by passive means, then within 30 minutes after formation of an initiation event the intervention of operating staff is not necessary. Any intervention of operating staff required or necessary within these 30 minutes shall in exceptional cases be duly substantiated. In the case that it is necessary to execute the required intervention separately or in combination outside the control rooms, such period shall be extended to 60 minutes.

3.15.1.5 ASSESSMENT OF INDIVIDUAL INITIATIING EVEVNTS

Reference: Safety Guide SÚJB BN-JB-1.0 (3), (22), WENRA App. E 7.1 - 7.5, IAEA SSR 2/1 5.2, 5.26, IAEA GS-R-4 Req. 16, 4.57

Acceptability of results of safety analyses shall be evaluated by their comparison with acceptance criteria. Based on nuclear power plant contractor design, these criteria shall be determined with the approval from the regulatory body so that the basic safety objective is fulfilled in an acceptable way - protection of population and the environment against adverse effects of ionising radiation in connection with operation of the power plant. With regard to their assumed frequency and severity of radiological consequences, radiation and technical design acceptance criteria shall be determined for each category of initiating events so that the initiating events with high frequency of occurrence have only insignificant radiological consequences and the events with severe radiological consequences have a low frequency of occurrence.

Acceptance criteria of safety analysis results shall be determined for all abnormal operation events and design basis accidents regarding the protection of the following:

- integrity of the fuel (i.e. the criterion for temperature/enthalpy of fuel, margin to boiling crisis, temperature of fuel covering, local and overall oxidation of coverage, admissible scope of damage of active zone, etc.)
- integrity of coolant pressure boundary (i.e. the criterion for maximum pressure and maximum temperature of coolant in the primary circuit and temperature and pressure changes and induced tensions in the pressure boundary of the primary circuit)
- integrity of the secondary circuit (i.e. the criterion for maximum pressure, maximum temperatures of media and temperature and pressure changes and tension in the equipment)
- integrity of the containment including the criteria for temperature and pressure of the environment including concentration of flammable/explosive gases and scope of release

The safety analysis shall check that radiological consequences of abnormal operation events and design basis accidents are in conformity with special legal procedure (Decree No. 307/2002 Coll. [L. 4]).

3.15.2 ANALYSES OF TRANSITION EVENTS AND DESIGN BASIS ACCIDENTS FOR THE NEEDS OF PRELIMINARY ASSESSMENT

This section includes the design characteristics in terms of compliance with the basic requirements for safety analyses for the purposes of the preliminary design concept assessment. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant design. This section includes partial assessment as to whether the preliminary concept of the design segment in question complies with requirements specified in Sections 3.15.1.1 to 3.15.1.5. The subject matter is the determination and assessment of general level of requirements for implementation of analyses of transition events and accidents whereas the particular method of implementation of these analyses shall be specified in the nuclear power plant design and evaluated in the following stage of safety documentation.

The basic requirements for the method of implementation of analyses of transition events and accidents is based on the general requirement for application of in-depth protection (in-depth protection requirements are specified in Section 3.3.1.1.3 and the requirements for the process of implementation of in-depth protection are specified in Section 3.3.1.1.4).

The method of implementation of analyses of transition periods and accidents shall meet at least the requirements specified in the binding legislation.

3.15.2.1 GENERAL PRINCIPLES AND OBJECTIVES OF ANALYSES IN THE PRELIMINARY DESIGN CONCEPT

A deterministic safety analysis shall be performed with the objective to verify that the power plant design is robust enough and balanced, that the actual power plant equipment has suitable properties, specifications and material composition, and that the pieces of equipment are suitably combined and placed in such a way that licensability in the Czech Republic is secured (see Section 1.5).

The analysis shall prove meeting of basic radiological acceptability criteria, i.e. that:

- radiation doses and discharges during normal and abnormal operation shall be as low as reasonably achievable and that they shall be under the levels stated in Section 3.12
- emergency radiation dosages for all design basis accidents shall be low and always below the values stated in Section 3.12

Acceptance criteria in terms of maintaining of physical barriers and efficiency of fulfilment of safety functions shall be also determined for the purposes of the analysis. At the equipment level, this means determination of acceptance criteria for the fuel (e.g. temperature and oxidation of coverage, fuel enthalpy, etc.) and individual systems (e.g. pressure, temperature, etc.) using internationally recognized standards. The criteria shall be graded depending on the frequency of occurrence of events so that more stringent criteria shall be applied to events with a higher frequency.

In terms of physical barriers it shall be demonstrated that

- the integrity of all physical barriers shall be preserved in all conditions of normal and abnormal operation
- the integrity of at least one barrier is maintained during design basis accidents

The deterministic analysis shall consider all postulated initiating events stipulated in line with the rules in Section 3.15.2.2, divided into categories according to occurrences frequency. Further it shall consider all external and internal impacts stated in Sections 3.3.3 to 3.3.5.

For each postulated initiating event or external or internal influence and their combinations, an analysis shall be implemented proving meeting of acceptance criteria. At the same time, the analysis proves that the safety systems are sufficiently sized and implemented in corresponding quality and qualification.

The analysis shall prove that the response of the power plant to any postulated initiating event shall meet the following requirements:

- the event shall not cause a substantial deviation from normal operation or does not cause a change to the safety state
- following a postulated initiating event, the plant shall be rendered safe by means of passive safety features or by the action of systems that are operating continuously in the state necessary to control the postulated initiating event
- following a postulated initiating event, the plant shall be rendered safe by the actuation of safety systems that need to be brought into operation in response to the postulated initiating event

It shall be proved that safety systems are able to transfer the power plant to a controlled status and subsequently transfer into and keep it in a safe status for a sufficiently long time. In terms of time, sufficient autonomy of safety systems shall be proved.

Calculation processes and programmes used in safety analyses shall be validated within the requisite scope to prove that they are able to creditably reproduce behaviour of real systems. The used data shall be subjected to demonstration that they are valid even in the given conditions using a reference to the acknowledged physical data, experimental data or in another way. In the case of subsequent use of several codes for analyses, the method of transfer of information between these codes shall be specified.

3.15.2.2 CLASSIFICATION OF TRANSIENT EVENTS AND ACCIDENTS IN THE PRELIMINARY DESIGN CONCEPT

Using deterministic and probabilistic methods and engineering judgement, identification of postulated initiating events shall be implemented. The selection of these postulated initiating events shall take into account all possible states of the power plant and all possible breakdowns of the equipment or operating staff errors including events at lowered power or with the reactor shut down. When selecting postulated initiating events, the content of radioactive material in the reactor shall be respected as well as in all other sources.

Events shall be divided into the limited number of categories, e.g. normal operation, abnormal operation, design basis accidents with low frequency, and design basis accidents with very low frequency. When dividing into categories, the frequency of the given initiating events shall be taken into account or their combination and the established licence practice, e.g. in the case of LB LOCA type events.

For each of the categories, a selection shall be made of the so-called cover events whose course and consequences cover the course and consequences of other accidents in the given category.

The following list shall be used for the selection of initiating events, which shall be in the following stage, i.e. in the initial safety report adjusted and complemented with events that are specific for the given design or site and which result from the results of probabilistic analysis.

Further the list from document WENRA [L. 27] Appendix E shall be used for selection of initiating events and internal influences as described in Section 3.15.1.2.

Normal operation

stable status, starting up, shutdown and shutdown status

- power operation
- start-up
- operation in hot reserve (no-power state)
- hot shutdown
- cold shutdown
- refuelling
- operation with shutdown loop, if permitted in the design

transient and non-standard states

- increasing and decreasing of temperature at maximum speed 55°C per hour
- step change of power (10 %)
- increasing and decreasing of power at 5 % per minute (between 15 % and 100 %)
- transition from power supply to service consumption (with steam bypassing)
- maximum values permitted according to operational limits and conditions

Abnormal operation

- inadvertent pulling of a group of control elements with subcritical reactor
- inadvertent pulling of a group of control elements in power state
- wrong position of control elements or falling of the group of control elements
- inadvertent thinning of primary coolant
- partial loss of flow via the primary circuit (e.g. outage of main circulating pump)
- inadvertent closing of a quick-acting valve on the main steam line
- loss of steam generator power supply
- failure of main steam generator power supply system
- complete loss of external power supply (shorter than 2 hours)
- excessive increasing of turbine power
- temporary depressurization of primary circuit (inadvertent activation of sprinkler system in volume compensator)
- inadvertent opening of safety valve on the steam line and other simple failures causing pressure drop in the secondary circuit
- inadvertent activation of emergency cooling system AZ
- failure in the system of replenishment, cleaning and recovery of coolant
- accidents with very low loss of primary coolant (small instrument line break)

Design basis accidents with low frequency of occurrence

- accidents with low loss of primary coolant (SB LOCA)
- accidents with low loss of secondary coolant
- decreasing of forced flow via active zone
- wrong feeding of fuel set
- inadvertent pulling of control body during power operation
- inadvertent activation of safety valve on volume compensator
- tank rupture in the system of replenishment, cleaning and recovery of coolant
- breaking of tank in the system of processing of gaseous waste
- breaking of tank in the system of processing of liquid waste
- breaking of one tube in the steam generator without prior leak of iodine from the fuel into primary circuit coolant
- complete loss of external power supply (shorter than 72 hours)

Design basis accidents with very low frequency of occurrence

- rupture of main steam line
- rupture of main feeding line
- seizure of rotor of main circulating pump
- shooting of one control body
- accidents with loss of primary coolant up to the size of guillotine leak from largest piping of primary circuit
- accidents when handling fuel
- breaking of one tube in the steam generator with prior leak of iodine from the fuel into primary circuit coolant

Further, identification of external and internal impacts shall be made with the objective:

- to prove that the probability of the given group of results with consequences more severe than the radiation acceptance criteria for design extension conditions is lower than 10-7/year
- to demonstrate that the consequences of the given impacts are in covering form contained in the already postulated initiating events
- to include external or internal impact into design premises

The consequences of external or internal impacts shall be generally regarded in combination with the most limiting conditions of normal operation whereas it is necessary to take into account:

- combination of various external and internal impacts which may occur at reasonable frequency, e.g. due to causal relation among them
- combination of various external impacts with postulated initiating events including shutdown which may occur at a reasonable frequency

- frequency and time of influence presence
- simple breakdown of systems necessary for transfer and maintenance of reactor in the state of safe shutdown (simple breakdown shall not have to be considered for impacts with very low frequency, e.g. plane crash)

For selection of internal and external impacts, the lists from Sections 3.3.3 through 3.3.5 shall be used which shall be in the following stage of licensing project, i.e. initial safety report adjusted and complemented with impacts which are specific for the given project or site and which result from the results of probability analysis.

3.15.2.3 POWER PLANT CHARACTERISTICS CONSIDERED IN SAFETY ANALYSES IN THE PRELIMINARY DESIGN CONCEPT

Characteristics of the power plant for the purposes of deterministic analyses shall be determined with sufficient design reserves and taking into account possible uncertainties.

The analysis shall take into account all subsequent failures formed as a result of initiating event and another independent simple failure shall be considered as well. For the events that deteriorate due to considered external power supply the loss of power supply shall be considered.

Further the impact of inoperability of the equipment due to testing and checks in operation shall be considered, unless it is possible to demonstrate sufficiently low frequency of concurrence of events requiring performance of the given safety function and its non-performance.

It shall not be necessary to consider a simple failure only for passive components that are designed, manufactured, mounted, checked, and maintained with a high level of quality provided that sufficient reasoning is given.

For the purposes of possible mitigation of consequences of accidents with the loss of primary coolant, the analysis may consider measures for piping break preclusion including leak before break.

Analyses shall be performed for simple independent initiating events. Simultaneous occurrence of an independent initiating event and external impact shall be considered only in the case that there is a causal relationship between them. Nevertheless, analysis of formation of other possible breakdowns during long-term restoration after a failure shall be performed.

The analysis shall consider only the contribution of safety systems within 24 hours of occurrence of the initiating event or achieving of a safe state (if it is achieved later than after 24 hours). Impact of other systems shall be considered only in the direction of deterioration of development of the given initiating event, unless it is possible to exclude the negative impact based on unchanging state of the given equipment. The contribution of the equipment associated with safety may not be considered until after 24 hours of occurrence of the initiating event, the contribution of non-safety systems then not until after 72 hours from the occurrence of the initiating event.

3.15.2.4 OPERATING STAFF INTERVENTIONS IN THE PRELIMINARY DESIGN CONCEPT

Safety analyses shall demonstrate that all considered operating staff interventions are feasible in terms of time and sufficient information provided.

An analysis shall be made of requisite interventions for performance of safety functions. It shall be checked that a sufficiently clear and simple system of displays including alarms is available in the operating and supplementary control room and in the technical support centre in the numbers sufficient for obtaining of the information regarding the state the power plant is in.

It shall be demonstrated that for meeting of radiation acceptance criteria for design basis accidents, no intervention from operating staff from the control room is necessary within 30 minutes of the initiating event. Further that intervention from operating staff outside the control room shall not be necessary within 60 minutes of the initiating event.

Analysis of human factor reliability shall be performed as a part of PSA.

3.15.2.5 ASSESSMENT OF INDIVIDUAL INITIATING EVENTS IN THE PRELIMINARY DESIGN CONCEPT

A deterministic safety analysis proving sufficient safety margin for acceptance criteria shall be implemented for each postulated initiating event (event category), or internal or external impact and their combination.

In addition to the standard conservative safety analysis of events of abnormal operation and design basis accidents, a realistic analysis including quantification of uncertainties shall be implemented at least for those selected design basis accidents from each group of events for which the difference between the conservative prediction and the acceptance criteria is the smallest . A sufficient safety margin shall be proven even in the cases when processes of evaluation of best approximation are used. If this margin is secured by means of conservative input data and other assumptions, then these assumptions shall be adjusted for the objectives of analyses for individual categories of events and specifically for individual applied criteria.

The acceptance criteria shall be assessed in terms of:

- damage to fuel (see Section 3.4.2.4),
- structural integrity, leaktightness and performance of functions of individual pieces of equipment according to their safety category,
- structural integrity and leaktightness of the system of containment,
- radiological consequences (see Section 3.12).

All relevant aspects of operational states and accident conditions (neutron-physical, thermo-hydraulic, structural, radiation) shall be analysed so that it is possible to evaluate performance of all acceptance criteria in a comprehensive way.

It shall be determined for each analysed initiating event which acceptance criteria are relevant and which physical parameters are the limiting ones. A separate selection of conservative initial and limiting conditions shall be made for each relevant acceptance criteria. If there is a variable possible selection of initial and limiting conditions for the considered type of event when verifying various acceptance criteria (whereas the impact of the conditions is not apparent), sensitivity calculations shall be made with the objective of appraising the impact of the given selection.

In the case of safety analyses it shall be taken into account that they are always burdened by uncertainties. In the cases when the uncertainties are significant in terms of the use of their results, sensitivity analyses shall be implemented and the uncertainties shall be quantified by sufficiently managed methods.

3.15.3 COMPREHENSIVE PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT

Analyses of transient conditions and design basis accidents shall meet the requirements based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and its implementation decrees for nuclear safety, radiation safety, emergency preparedness and also requirements at IAEA SSR 2/1 [L. 252] and IAEA GS-R-4 [L. 278] and WENRA [L. 27] and [L. 270].

The method of stipulation of requirements for safety analyses and the said assessment confirm that the assumed method of implementation of analyses of transient conditions and accidents creates prerequisites for meeting the relevant requirements stipulated by Decree No . 195/1999 Coll. [L. 266], Safety Guide SÚJB BN-JB-1.0 [L. 277], document IAEA SSR 2/1 [L. 252], IAEA GS-R-4 [L. 278] and document WENRA [L. 27].

Specification of requirements for implementation of analyses of transient conditions and accidents characterized in Section 3.15.2 was created in conformity with the licensee's requirements applied to potential nuclear facility contractors within the tender and creates the concept of solution of this design segment.

The particular method of implementation of analyses of transient conditions and accidents shall be specified in detail in the nuclear power plant design.

3.16 OPERATIONAL LIMITS AND CONDITIONS

This comprehensive part of the ISAR is structured into three basic sections:

The introductory Section 3.16.1 summarizes, analyses, and specifies basic legislation requirements for operational limits and conditions including determination of their objective, scope, way of making changes and delimitation of application.

The following Section 3.16.2 contains a description and determination of basic requirements for operational limits and conditions within all projects that may be considered within the tender procedure under way. The objective of the section is to formulate the design's general characteristics for the purposes of partial preliminary assessment.

The final Section 3.16.3 contains a comprehensive preliminary assessment of the design concept in terms of operational limits and conditions, summarising the conclusions of the partial preliminary assessments presented in Section 3.16.2. The assessment of the summarized requirements contains a preliminary design concept assessment required by the law.

In the following stage of the licensing documentation, the applicant shall provide evaluation and support information which shall allow assessment that the submitted part of the ISAR contains adequate scope of specified operational limits and conditions including the data on permissible parameters, setting of protection systems, operability conditions and requirements for power plant staff activities to secure safe operation within design basis conditions. The section shall also be supplemented with detailed information at the depth and in the structure described in RG 1.206 [L. 275].

3.16.1 BASIC LEGISLATION REQUIREMENTS FOR OPERATIONAL LIMITS AND CONDITIONS (OLC)

3.16.1.1 OBJECTIVE OF OPERATIONAL LIMITS AND CONDITIONS (OLC)

Reference: WEN RA Issue H 1.1, 1.2

OLCs shall be developed as a covering comprehensive document to ensure that plants are operated in accordance with design assumptions and intentions as documented in the respective safety analysis report. They shall define the conditions that must be met to prevent situations that might lead to accidents or to mitigate the consequences of accidents should they occur.

3.16.1.2 FORMATION AND CHANGES OF OLC

Reference: IAEA SSR 2/1 Req. 28, WENRA Issue H 2.1, Safety Guide SÚJB BN-JB-1.0 (48)

In the course of the designing stage a list of requirements and limits shall be formed for the operation of the nuclear facility of which OLCs shall be developed for safe operation of the power plant. Every created limit shall be reasoned by the design, safety analysis, and start-up tests.

Reference: WENRA Issue H 2.2, 2.3

OLCs shall be kept updated and reviewed in the light of experience, developments in science and technology, and every time modifications in the plant or in the safety analysis warrant it, and changed if necessary.

The process for making modifications or temporary modifications of OLCs shall be defined. Such modifications shall be adequately justified by safety analysis and independent safety review

Reference: Decree No. 195/1999 Article 4 (3), WENRA Issue O 3.4

Probability risk assessment shall be used for evaluation of suitability of changes to operational limits and conditions. The quality and suitability of computational programs used for the analysis important for nuclear safety shall be verified.

3.16.1.3 SCOPE AND CONTENT OF OLCS

Reference: WENRA Issue H 4.1, Safety Guide SÚJB BN-JB-1.0 (48)

OLCs shall cover all operational states of the power plant including power operation, shuttdown, refuelling, all transient forms between these conditions and temporary conditions which occur due to repairs and tests.

Reference: IAEA SSR 2/1 Req.28 5.44, WENRA Issue H 5.1, 5.2, 6.1, 6.2, 6.3, 9.1, Safety Guide SÚJB BN-JB-1.0 (48)

OLCs shall contain:

1. Safety limits

A safety limit is the critical value of a certain operational parameter which prevents failure of the equipment designed for protection against uncontrolled releasing of radioactive substances.

Safety limits shall be determined using a conservative approach taking into account uncertainties in safety analyses.

2. Setting of safety systems

Via adequate differences between operational limits and specified setting of safety systems it shall be secured that frequent undesirable activation of safety systems does not occur.

3. Limit conditions for operation

OLCs for normal operation shall contain limits for operational parameters, setting of minimum numbers of operational pieces of equipment, activities which operational staff have to perform in the case of deviation from OLCs and deadlines for completion of these activities.

If the requirements for operability may not be met, limits shall require activities securing transition of the power plant into a more safe state, whereas deadlines for completion of these activities have to be stated.

Requirements for operability of various types of normal operation shall be stated by the number of the systems or components important to safety which shall be in operating or standby condition and limitations in the case they are not available.

4. Limitation of operational or other important parameters by the control system or administrative process.

5. Requirements for surveillance, maintenance, tests, inspections and testing to prove that systems, structures and components work as specified by the design whereas

- ALARA principles shall be respected
- specified frequency of checks and tests shall be sufficient for verification of reliability and at the same time they shall not cause to degradation of the equipment and unfounded shortening of lifetime
- the results shall be evaluated and archived

6. Directions for activities as reactions to deviations from specified OLCs including the time for performance of these activities.

Reference: WENRA Issue H 7.1, 7.2, 8.1, 10.1, 10.2

If operating personnel cannot ascertain that the power plant is operating within operating limits, or the plant behaves in an unexpected way, measures shall be taken without delay to bring the plant to a safe and stable state.

The plant shall not be returned to service following unplanned shutdown until it has been shown to be safe to do so.

Minimum staffing levels for shift staff shall be stated.

In the case of breaching of OLCs, remedial measures shall be adopted in order to immediately renew performance of OLCs. The reports of breaching of OLCs shall be investigated and remedial measures shall be taken to prevent recurrence of such breaching in the future.

3.16.1.4 OBSERVANCE OF OLCS

Reference: WENRA Issue H 3.1, 3.2, Safety Guide SÚJB BN-JB-1.0 (95)

The OLCs shall be readily accessible to control room personnel.

Control room operators shall be highly knowledgeable of the OLCs and their technical basis. Relevant operational decision makers shall be aware of the significance of OLCs for the safety of the plant.

3.16.2 PROPERTIES OF THE OLC DESIGN FOR THE NEEDS OF THE PRELIMINARY ASSESSMENT

This section describes the properties of the design for the needs of a preliminary assessment of the design concept. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant design. This section includes partial assessment as to whether the preliminary concept of the design segment in question complies with requirements specified in Sections 3.16.1.1 to 3.16.1.4. The subject-matter is the determination and assessment of the general level of requirements for the design in terms of operational limits and conditions whereas the particular method of application of these requirements shall be

specified in the power plant design and evaluated in the following stage of safety documentation.

Objective of operational limits and conditions (OLC)

OLCs shall contain a file of unambiguously defined conditions proving that when these are performed, the power plant operation shall be safe. They shall contain data on admissible parameters, equipment operability requirements, setting of protection systems, requirements for activity of workers and organization of measures to meet all defined conditions for the designed operating conditions.

OLCs shall form a document whose performance shall prove that the power plant is operated in conformity with design assumptions and intentions. Performance of the OLCs shall prevent situations that might lead to accidents or mitigate the consequences of accidents should they occur.

Formation and changes of OLCs

In the designing stage an OLC file shall be created. When forming OLCs, the contractor shall use the analysis of previously occurred relevant events in the similar equipment.

When creating OLCs, probability assessment of safety shall be used and it shall be based on the design. Proposed OLCs shall be a part of the documentation for testing.

OLCs shall be a part of the contractor's documentation for operation and maintenance. The OLCs including their reasoning shall be approved by SÚJB.

Changes of OLCs:

- changes shall be performed in line with an approved procedure
- they shall be implemented with knowledge of information on design margins in relation to OLCs which shall be provided by a nuclear facility contractor
- every change shall be sufficiently reasoned by safety analysis, independent safety assessment and probabilistic evaluation of safety
- they shall be implemented based on experience, development of science and technology, and if necessary due to development of the power plant

Content of OLCs

The OLC file from the designing stage shall be used as the basis for the OLCs and it shall contain the following:

- setting of safety systems
- limitation of operational or other important parameters by the control system or administrative process
- recommendation for maintenance, testing and inspections of the power plant to verify that systems, structures and components work according to the design including ALARA
- clearly defined operational configuration including operational limitations in the case of outage/shutdown of safety systems

Use of OLCs

In order to facilitate use of OLCs, control systems shall be designed so as to provide computer support to the operating staff. Control systems shall notify power plant staff of approximation or breach of operational limiting conditions in all operational states of the power plant.

When training on a simulator, the administrative processes which closely cooperate with control systems and are used in power plant operation shall be simulated in the scope relevant for achieving of training goals. Particularly the systems monitoring compliance with the OLCs and safe state of the power plant shall be included in the scope of simulation.

Partial preliminary assessment

The preliminary concept of the design summarizing the most important requirements for the design in terms of operational limits and conditions stated in Section 3.16.2 creates prerequisites for meeting the requirements of Decree No. 195/1999 [L. 266] Article 4 (3), IAEA SSR 2/1 [L. 252] Req. 28, Req.28 5.44, Safety Guide SÚJB BN-JB-1.0 [L. 276] (48), (95) and WENRA [L. 27] Issue H 1.1, 1.2, 2.1, 2.2, 2.3, 3.1, 3.2, 4.1, 5.1, 5.2, 6.1, 6.2, 6.3, 7.1, 7.2, 8.1, 9.1, 10.1, 10.2, Issue O 3.4.

3.16.3 PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT

The ETE 3,4 design shall meet the safety requirements specified in Section 3.3.1.1 as well as the requirements specified in Sections 3.16.1.1 to 3.16.1.4, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and related implementing decrees as regards nuclear safety, radiation protection and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270].

The principles of the design solution described in Section 3.16.2, were set up based on the requirements submitted by the applicant for licence applied on the potential contractors of the nuclear installation within the tender, and it creates the concept of the design solution for this part of the design. The implemented partial assessments confirm that the requirements applied to the design in terms of operational limits and conditions create prerequisites for fulfilment of relevant requirements specified by Decree No. 195/1999 Coll. [L. 266], Safety Guide SÚJB BN-JB-1.0 [L. 276], IAEA SSR 2/1 [L. 252] and WENRA [L. 27] documents.

The particular method of fulfilment of individual requirements for the operational limits and conditions system stated in Section 3.16.2 shall be drawn up in detail in the PSAR drawn up based on the selected project.

3.17 QUALITY ASSURANCE

This comprehensive part of the ISAR in its introductory Section 3.17.1 summarizes, analyses, and specifies basic legislation requirements for quality assurance in the course of designing, construction, and operation of the nuclear installation. The following Section 3.17.2 contains a description and determination of basic requirements for quality assurance specified in the previous section forming the design basis requirements and is intended for stipulation of the characteristic of the design for the needs of preliminary assessment. The final Section 3.17.3 contains a comprehensive assessment of quality assurance design concept.

In the following stage of the licence documentation, the applicant shall provide evaluating and supporting information, allowing assessment of quality assurance throughout the entire reactor unit lifetime in all modes within the design requirements, including steady-state and transient operation regimes, also during the occurrence of design extension conditions.

3.17.1 BASIC LEGISLATIVE REQUIREMENTS FOR PROVISION OF QUALITY ASSURANCE

3.17.1.1 REQUIREMENTS FOR QUALITY ASSURANCE DURING DESIGNING AND CONSTRUCTION

This chapter summarizes basic requirements for quality assurance during designing and construction. The details of the documentation drawn up for demonstration of quality assurance towards a surveillance body are described in Section 6 of this ISAR.

Reference: SÚJB Safety Guide BN-JB-1.0 (14)

Prior to the start of the process of designing the nuclear installation and in its course, which is according to Act No. 18/1997 Coll. [L. 2] one of the activities related to the utilization of nuclear energy, a quality assurance programme shall be approved for designing and a respective quality assurance system shall be established according to a special decrees so that the equipment important to nuclear safety and radiation safety are designed (manufactured, erected, and tested) in a quality corresponding to their safety significance for safe provision of safety functions.

Reference: SÚJB Safety Guide BN-JB-1.0 (15); IAEA SSR 2/1 Req. 2

An integrated management system shall be established applying the requirements of quality assurance system, which shall secure that any activities, intentions, and requirements shall not be dealt with individually or separately from safety requirements. The relations and responsibilities of individual participants in the designing process and related processes shall be unambiguously defined.

Reference: SÚJB Safety Guide BN-JB-1.0 (16)

The organization designing the nuclear installation shall have sufficient knowledge and experience to know and understand all physical occurrences and properties of individual used pieces of equipment and characteristics of the nuclear installation as a whole and shall be able to consider all states in which they shall or may be including associated risks.

Reference: SÚJB Safety Guide BN-JB-1.0 (17)

For the activities during designing in the period before commissioning the nuclear installation, a professionally competent person (organization) shall be formally established as responsible for the whole process of designing, and securing of design integrity in the course of construction and commissioning.

Reference: IAEA SSR 2/1 Req. 1

The applicant for permission to build or operate a nuclear power plant shall submit to the supervisory body a draft that shall complement all effective safety requirements.

Reference: IAEA SSR 2/1 3.1

All organizations, including the design organization, engaged in activities important to the safety of the design of a nuclear power plant shall be responsible for ensuring that safety matters are given the highest priority.

Reference: IAEA SSR 2/1 3.2

The management system shall include a measure for ensuring the quality of the design of each structure, system and component, as well as of the overall design of the nuclear power plant, at all times. This includes the means for identifying and correcting design deficiencies, for checking the adequacy of the design and for controlling design changes.

Reference: IAEA SSR 2/1 3.3

The design of the plant, including subsequent changes, modifications or safety improvements, shall be in accordance with established procedures that call on appropriate engineering codes and standards and shall incorporate relevant requirements and design bases. Interfaces shall be identified and controlled.

Reference: IAEA SSR 2/1 3.6 (2, 5)

The design organization shall ensure that the plant design meets the acceptance criteria for safety, reliability and quality in accordance with relevant national and international codes and standards, acts and decrees. A series of tasks and functions shall be established and implemented to ensure the following:

(2) the design verification, definition of engineering codes and standards and requirements, use of proven engineering practices, provision for feedback of information on construction and experience, approval of key engineering documents, conducting of safety assessments and maintaining a safety culture are included in the function;

(5) the necessary interfaces with responsible designers and suppliers engaged in design work are established and controlled.

Reference: WENRA NEW O7

The licensee establishes effective management for safety over the entire new plant project and secures sufficient in-house technical and financial resources to fulfil safety requirements. The licensee ensures that all other organizations involved in the project demonstrate awareness among the staff of the nuclear safety issues associated with their work and their role in ensuring safety.

3.17.1.2 QUALITY ASSURANCE REQUIREMENTS DURING OPERATION

This section summarizes basic requirements for quality assurance during operation of the nuclear installation. The details of the documentation drawn up for demonstration of quality assurance towards a surveillance body are described in Section 6 of this ISAR.

Reference: SÚJB Safety Guide BN-JB-1.0 (18)

The licensee shall establish in its management system a corresponding measure in order to secure sufficient knowledge of all aspects of the project necessary for securing of nuclear safety and radiation safety during operation and in the case of changes of the project and operating conditions. These measures shall include, among other things, determination of an organization unit responsible for knowledge and safety of the project and securing of access to information of the original author of the design and the supplier of the equipment.

Reference: WENRA Issue B 2.1

The licensee shall secure that the equipment shall be operated in a safe manner and in conformity with effective legal requirements and the requirements of the supervisory body.

3.17.1.3 REQUIREMENTS FOR QUALITY ASSURANCE PROGRAMME DESCRIPTION

The quality assurance programme of the applicant for licence shall be stated in Section 6.1 of this ISAR.

3.17.1.4 REQUIREMENTS FOR MANAGEMENT OF RELIABILITY ASSURANCE PROGRAMME

This shall be specified within the next stages of the licensing documentation.

3.17.1.5 REQUIREMENTS FOR QUALITY ASSURANCE PROGRAMME MANAGEMENT

This chapter summarizes basic requirements for activities when managing the quality assurance programme. The process of issuing and the content of programmes is described in Chapter 6 of this ISAR.

Reference: IAEA SSR 2/1 3.6 (3), WENRA Issue B 2.5

The applicant for licence shall ensure that continuous and systematic use of operational experience, international development in the area of safety standards and new findings in research is secured for safe operation, maintenance and improvement of the installation.

Reference: IAEA SSR 2/1 3.6 (4, 6, 7)

The applicant for a licence shall secure that management of design requirements shall be set and checked. Further, the applicant shall secure that the necessary engineering judgement and scientific and technical knowledge are maintained within the operating organization and that all design changes to the plant are reviewed, verified, documented and approved.

Reference: WENRA Issue B 2.2

The applicant shall ensure that decisions on safety matters are preceded by appropriate investigation and consultation so that all relevant safety aspects are considered. Safety issues shall be subjected to appropriate safety review, by a suitably qualified independent review function.

Reference: WENRA Issue B 2.4

The applicant shall ensure that safety performance is continuously monitored through an appropriate review system in order to ensure that safety is maintained and improved as needed.

Reference: WENRA Issue B 2.6

The applicant shall ensure that plant activities and processes are controlled through a documented quality management system covering all activities, including relevant activities of vendors and contractors, which may affect the safe operation of the plant.

3.17.1.6 REQUIREMENTS FOR MAINTENANCE PROGRAMME DESCRIPTION

Reference: WENRA Issue B 2.3

The applicant shall ensure that the staff is provided with the necessary facilities and working conditions to carry out work in a safe manner.

3.17.2 PROPERTIES OF THE QUALITY ASSURANCE PROJECT FOR THE NEEDS OF THE PRELIMINARY ASSESSMENT

This section describes the properties of the design for the needs of a preliminary assessment of the design concept. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant design. This section includes partial assessment as to whether the preliminary concept of the design segment in question complies with requirements specified in Sections 3.17.1.1 to 3.17.1.6. The subject matter is the determination and assessment of general level of requirements for quality assurance, whereas the particular method of quality assurance shall be specified in the nuclear power plant design and evaluated in the following stage of safety documentation.

The nuclear installation contractor shall create a Quality and Environmental General Plan (QAEGP) in which the contractor describes the quality assurance system throughout the whole project. QAEGP is a top quality assurance document approved by the licensee.

Based on the approved QAEGP, processes and quality plans shall be prepared for each project stage (siting, designing, manufacture, construction, commissioning, operation), which shall also be submitted to the applicant for approval or for information. Individual activities shall not be started without the respective process or plan. All these documents shall be used as supporting materials for drawing up QAPs for individual stages submitted to SÚJB for approval.

QAEGP shall be in conformity with acts, decrees, codes and standards in the following order:

- Atomic Act No. 18/1997 Coll. [L. 2], Decree No. 132/2008 Coll. [L. 258]

- IAEA Safety Fundamentals No. SF-1 „Fundamentals Safety Principles“, IAEA General Safety Requirements No. GS-R-3 "The Management System for Facilities and Activities" and other related IAEA standards from Safety Fundamentals and General Safety Requirements category; further, European Standard EN ISO 9001 "Quality Management Systems – Requirements" and EN ISO 14001 "Environmental Management Systems, Requirements with guidance for use"
- The acts, codes and standards used for the design in the country of origin or in the European Union country where the project is licensed
- IAEA General Safety Guide No. GS-G-3.1 „Application of the Management System for Facilities and Activities“ and IAEA Specific Safety Guide No. GS-G-3.5 „The Management System for Nuclear Installations“
- ISO standards, ISO 10005 "Quality management systems – Guidelines for quality plans", ISO 10006 "Quality management systems – Guidelines for project quality management" and ISO 10007 "Quality management systems – Guidelines for configuration management"

The contractor shall secure that each subcontractor issues a Quality Assurance and Environmental Plan (QAEP), which shall be in conformity with QAEGP requirements. The contractor shall be responsible for monitoring and evaluation of QAEGP requirements in subcontractors' QAEPs. All plans and quality processes of a subcontractor shall be approved by the contractor and they shall be submitted to the applicant for approval or for information.

Quality assurance requirements shall take into account the approach described in IAEA General Safety Requirements document No. GS-R-3 "The Management System for Facilities and Activities" and they shall be set with regard to safety functions of individual systems, structures and components (SSC).

The importance of processes for quality assurance shall be set so as to reflect the significance of activities and classification of equipment. Classification of the equipment shall be in conformity with Decree No. 132/2008 Coll. [L. 258].

The contractor is responsible for setting of the method of evaluation of efficiency of quality assurance system and environmental protection.

Software, special procedures, and workers performing functions associated with safety or operability of the power plant must be verified or must pass through a qualification process.

The contractor and/or subcontractor shall secure that systems, structures and components always meet specifications stated in the contractual stipulations and in power plant design requirements.

The contractor shall pay special attention to design checking and verification of all processes. The licence applicant or applicant's representative shall have the right to perform monitoring and verification of design and all processes on their own in order to secure that the contractual and design requirements are thoroughly performed.

All requirements resulting from design activities shall be monitored and evaluated in the course of the whole project. Changes of individual SSCs compared to the original requirements shall be subject to a controlled system of changes which shall be defined. The persons that are to monitor all changes and that shall propose, verify, and approve changes of individual SSCs shall have requisite expert knowledge.

All requirements with effect on safety and reliability of individual SSCs shall be continuously monitored so that their functions and significance remain preserved or possibly are continuously improved. Safety and reliability issues shall have the highest priority. Decisions regarding these questions shall be preceded by qualified and independent analyses and verification proving that the result of the decision shall not cause decreasing or loss of their significance.

In the course of the whole project the applicant for the licence shall secure sufficient technical and financial resources. All organizations involved in the design shall be aware of safety issues and their role in securing of safety.

The applicant for licence shall secure sufficient knowledge of all aspects of the design in order to secure nuclear safety and radiation safety during operation and in the case of changes of the design and operational conditions.

The applicant for licence, applicant's representative, and the regulatory body shall have the right to monitor, perform audits and checks to verify knowledge and experience of the contractor and the organizations participating in the design. The auditing programme shall be an integral part of the approved QAEGP.

QAEGP shall contain a description of documentation checks. The applicant for licence or applicant's representative shall have the right of access to all quality documents created in the course of the project.

QAEGP shall contain all procedures for settlement of nonconformities and design changes in particular with regard to the following conditions:

- The adopted remedial measures and implemented design changes shall not decrease safety and reliability of the power plant
- The cause of nonconformity shall be ascertained and the remedial measure shall prevent recurrence of similar nonconformities

The processes for dealing with nonconformities shall have a specified information flow and instances shall be described when the applicant's approval is required for dealing with a nonconformity.

QAEGP shall contain a list of all quality records used during the project. Handover of records of quality assurance shall be implemented by means of the DMS (Documentation Management System).

QAEGP shall contain a description of archiving of all quality records (particularly time and form of archiving). All records shall be protected against damage or destruction.

The applicant for the licence shall have access to information regarding the stage of processing of individual records (preparation, commenting, approving, issuing). The records shall:

- be traceable,
- have correctly recorded results of inspections, checks and tests
- have clear assessment that SSC, project stages and individual activities to which the records relate are in conformity with the defined criteria

Special attention shall be paid to the interface between individual organizations cooperating within the project. All interfaces shall be identified, defined and updated in all stages of the design. Transfer of records between organizations cooperating within the design and traceability of information related to project quality assurance shall be accessible to the applicant for the licence upon request.

Records of quality assurance shall be provided by all organizations participating in the design. All records of systems, structures, and components (SSC) and individual stages of the project shall be presented to the applicant for the licence for approval or for information.

All employees implementing activities in the course of the project shall have suitable work and protection equipment. The applicant for the licence shall require, monitor, and continuously evaluate the use of protective and work equipment in order to secure work safety. The working environment too shall meet all requirements important to safety and occupational health protection.

Partial preliminary assessment

The preliminary design concept summarizing the most important requirements for quality assurance stated in Section 3.17.2 creates prerequisites for meeting the requirements of document IAEA SSR 2/1 [L. 252] Req. 1, Req. 2, 3.1-3.4, 3.6 (2, 3, 4, 5 6, 7), Safety Guide SÚJB BN-JB-1.0 [L. 276] (14-17) and document WENRA [L. 27] Issue B 2.1 – 2.6., WENRA NEW O7 [L. 270].

3.17.3 COMPREHENSIVE PRELIMINARY ASSESSMENT OF THE QUALITY ASSURANCE DESIGN CONCEPT

The ETE 3,4 design shall meet the safety requirements specified in Section 3.3.1.1 as well as the requirements specified in Sections 3.17.1.1 to 3.17.1.6, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and related implementing decrees as regards nuclear safety, radiation protection and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270].

To meet those requirements, the ETE 3,4 design shall use systems, structures and components and their equipment integrated within safety systems such as shall ensure (with adequate reliability and resistance) qualitatively and quantitatively adequate safety and technological functions in accordance with the specified requirements for the prescribed safety functions.

The principles of the design solution described in Section 3.17.2 were set up based on the requirements submitted by the applicant for licence applied on the potential suppliers of the nuclear installation within the tender, and they make up the concept of the design solution for this part of the design. The implemented partial assessments confirm that the assumed design solutions of quality assurance meet the specific requirements for systems, structures and components, safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], Safety Guide SÚJB BN-JB-1.0 [L. 276], IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and WENRA NEW [L. 270] documents. Quality assurance complies with the implemented preliminary assessment of the design concept.

The particular method of implementation of individual requirements stated in Section 3.17.2 shall be specified in detail in the nuclear power plant design.

3.18 HUMAN FACTORS ENGINEERING

This comprehensive part of the ISAR is structured into three basic sections:

The introduction Section 3.18.1 summarizes, analyses, and specifies basic requirements for the area of human factors engineering including the requirements for planning and managing of the approach to engineering psychology ergonomics, designing of control rooms and requirements for staff necessary for their operation. The section also includes requirements for verification and demonstration of the implemented design and the requirements for the stage of implementation of design and operation.

The following section 3.18.2 contains a description and determination of basic requirements for functions and principles within human factors engineering specified in the opening Section 3.18.1 in the form forming an envelope of the applied design requirements for functional units, systems and activities relevant for the area of human factors engineering within all projects that may be considered within the current tender. The objective of the section is to formulate the design's general characteristics for the purposes of partial preliminary assessment.

The final Section 3.18.3 contains a comprehensive preliminary assessment of the concept of human factors engineering design that summarizes the conclusions of the partial preliminary assessments completed in Section 3.18.2. The assessment of the summarized requirements contains a preliminary design concept assessment required by the law. The applicant shall use the next section of the licence documentation in order to provide assessment and supporting information on the selected design that shall allow assessment of the ability of the drafted design within human factors engineering to fulfil the specified safety functions during the nuclear unit's lifetime in any operating conditions, including accident conditions. The section shall also be supplemented with detailed information at the depth and in the structure described in RG 1.206 [L. 275].

3.18.1 BASIC LEGISLATION REQUIREMENTS FOR HUMAN FACTORS ENGINEERING

Human factors engineering is the sum of knowledge applied to designing human-machine interfaces. In terms of complexity of many integrated systems, the application of human factors engineering has a substantial impact on the design and thus on reliable operation of the nuclear power plant.

The objective of human factors engineering is to reflect the results of implemented functional analyses into the design and optimally integrate human, technical and other criteria in order to meet the prerequisites for achieving of safety and operational objectives of the power plant.

This particularly includes accessibility of accurate and timely information suitably provided in relation to the operational situation and further decreasing of the workload of operating staff.

The main factor when designing and assessing the design are the skills and qualities of the operating staff in terms of physical and mental abilities and skills. This systematic approach also includes securing of the working environment and protection of health of operating staff in control rooms.

3.18.1.1 REQUIREMENTS FOR PLANNING AND MANAGEMENT OF THE APPROACH TO HUMAN FACTORS ENGINEERING

The section contains basic requirements for planning and management of approach to human factors engineering divided into three areas:

- Systematic approach to the design
- Reflection of results of functional analyses
- Man-machine interface

Systematic approach to the design

Reference: IAEA SSR 2/1 Req. 32

Systematic consideration of engineering psychology-ergonomics including the man-machine interface, shall be included at an early stage in the design process for a nuclear power plant and shall be continued throughout the entire design process.

Reference: Safety Guide SÚJB BN-JB-1.0 (95), IAEA SSR 2/1 Req. 32, 5.54, 5.55, 5.58, 5.59, 5.61

Operating personnel who have gained operating experience in similar plants shall, as far as is practicable, be actively involved in the design process conducted by the design organization, in order to ensure that consideration is given as early as possible in the process to the future operation and maintenance of equipment.

The design shall support operating personnel in the fulfilment of their responsibilities and in the performance of their tasks, and shall limit the effects of operating errors on safety. The design shall be such as to promote the success of operator actions with due regard for the time available for action, the conditions to be expected and the psychological demands being made on the operator.

The design process shall pay attention to plant layout and equipment layout, and to procedures, including procedures for maintenance and inspection, to facilitate interaction between the operating personnel and the plant.

The design of workplaces and the working environment of the operating personnel shall be in accordance with ergonomic concepts.

The need for intervention by the operator on a short time scale shall be kept to a minimum, and it shall be demonstrated that the operator has sufficient time to make a decision and sufficient time to act. The information necessary for the operator to make a decision to act shall be simply and unambiguously presented.

Reference: IAEA SSR 2/1 Req. 32, 5.62

Verification and validation (including by simulators) of features relating to human factors shall be included at appropriate stages to confirm that necessary actions by the operator have been identified and can be correctly performed.

Reflection of results of functional analyses

Reference: Decree No. 195/1999 Coll., Article 4 (1); SÚJB BN-JB-1.0 Safety Guide (83), IAEA SSR 2/1, Req. 59, 60, 6.31

The nuclear power plant shall be equipped with control and information systems allowing monitoring, measuring, recording, further processing, and control of the operating parameters, technological processes, and systems essential for securing nuclear safety and radiation and physical protection, and accident preparedness during normal and abnormal operation and in accident conditions. The control and information systems shall provide the necessary visual and audio warnings informing of new or changed operating states, processes, and parameters that deviate from the admissible limits for the normal operation and may affect safety.

The control systems shall secure maintaining of the operating parameters of supporting equipment that is important to nuclear safety in line with the acceptance criteria, limits, and safe operating conditions.

The requirements for instrumentation and control systems are stated in Section "3.7.1 The basic legislative requirements for the instrumentation and control systems".

The operating staff shall have available sufficient information the scope and presentation form of which shall result from functional analyses. This information shall allow immediate evaluation of the power plant state whatever the plant's conditions may be and confirm implementation of proposed automatic safety activities.

Human-machine interface

Reference: Decree No. 195/1999 Coll., Article 16 (2, 3), SÚJB BN-JB-1.0 Safety Guide (83, 84, 95), IAEA SSR 2/1 Req. 32, 5.56, 5.57, 5.61

Communicators of status of parameters and equipment controllers shall be designed and distributed in order that human factor and ergonomic requirements for man-machine interface are respected and that the operating staff always have enough easily manageable information on the operation of the nuclear installation, automatic interventions of controlling protection systems and their results and may intervene as necessary.

The control and information systems shall provide the necessary visual and audio warnings informing of new or changed operating states, processes, and parameters that deviate from the admissible limits for the normal operation and may affect safety.

The control and information systems will record continuously in regular intervals, or as needed, the values of the parameters that are necessary for nuclear safety of the nuclear installation in accordance with safety analyses.

In the case of accident conditions, the instrumentation shall provide:

- information regarding the current state of the nuclear installation, based on which the protective measures can be carried out. This information shall include at least information on parameters and system states, which can affect the course of the fission reaction, integrity of the active zone, integrity of the primary circuit and the containment and related systems
- basic information on the course of the accident and its recordings
- The information that enables the prognosis of the spread of radionuclides and ionising radiation to the vicinity of the nuclear power plant so that it would be possible to implement timely measures for the protection of the population.

3.18.1.2 REQUIREMENTS FOR DESIGNING - CONTROL ROOMS AND SUPPORT CENTRES

Control room

Reference: Decree No. 195/1999 Coll., Article 20 (1, 2), SÚJB BN-JB-1.0 Safety Guide (93, 94, 95), IAEA SSR 2/1 Req. 32, 65, 5.60, 5.61, 6.39, 6.40, WENRA App. E 10.3, 10.4, 10.5

The nuclear power plant shall be equipped with at least one control room, from which the power plant may be operated in all operating states either automatically or manually by an operator. From the control room measures to bring the plant to a safe shutdown state shall be taken following the appearance of expected operational events and design basis accidents.

Special attention shall be paid to the determination of internal and external events and subsequent accident conditions that may present a direct threat to further operation of the main and supplementary control room and the design shall secure reasonably feasible measures for minimization of the impact of these events.

When designing the control room, ergonomic approaches shall be respected.

The control room shall be designed in such a way so that in terms of operating staff protection it allows access, safe staying and a harmless environment in the control room even in accident conditions. It means that suitable measures shall be taken including establishment of barriers between the control room and external environment and respective information shall be provided for control room operating staff in order to protect them against danger (high radiation level as a result of accident conditions, releasing of radioactive materials, fire, explosions, release of poisonous gases).

Information systems of the control room shall secure by means of visual or sound indications that the operating staff is notified in due time of deviations from operational states and processes that may affect nuclear safety. Operators shall have available corresponding information for monitoring results of automatic interventions.

Supplementary control room

Reference: Decree No. 195/1999 Coll., Article 20 (3), Safety Guide SÚJB BN-JB-1.0 (95, 96), IAEA SSR 2/1 Req. 32, 66, 5.60, 5.61, 6.41, WENRA App. E 10.6

The design shall be designed in a manner enabling shutdown, maintenance of the reactor in the safe state, and removal of residual heat, as well as monitoring of the power plant's state even if the control room becomes unusable. The design shall contain the respective back-up system that may have the nature of a supplementary control room and shall be sufficiently physically and electrically separated from the control room.

When designing the supplementary control room, ergonomic approaches shall be respected.

Requirements for habitability of the control room, as well as the supplementary facility, are specified in Section "3.6.1.4 Habitability systems".

Emergency support centres

Reference: Safety Instruction SÚJB BN-JB-1.0 (97), IAEA SSR 2/1 Req. 67, 6.42

An on-site technical support centre, separate from both the plant control room and the supplementary control room, shall be provided from which an emergency response can be directed at the nuclear power plant.

Information shall be transmitted to this technical support centre and displayed on important power plant parameters available to the control room operating staff in order to allow timely evaluation of the state of the nuclear installation and critical safety functions in accident conditions by members of the technical support centre active in implementation of organizational measures for management of radiation accidents and accidents within power plant premises and its immediate vicinity.

The work station shall be equipped with communication means for connecting with the control room, supplementary control room and other important places within the power plant and other work stations of accident response within the power plant and outside it. The emergency control centre shall include the necessary systems and services to permit extended periods of occupation and operation by emergency response personnel.

The requirements for emergency support centres are stated in Section "3.13.1.3 The requirements for planning of activities in emergency situations or 3.3.1.1.17 Means of Communication".

Staff, training equipment and programmes

Reference: IAEA SSR 2/1 Req. 32, 5.53, WENRA Issue B 3.1, 3.2

The design shall specify the minimum number of operating staff members necessary for performance of all simultaneous activities necessary for bringing the plant to a safe shutdown state.

The required number of staff members for securing of safe operation and their competencies and suitability for safety work shall be analysed in a systematic and documented way.

Reference: WENRA Issue D, 3.1, 3.3, 3.4

Performance based training programmes shall be established for all staff with tasks important to safety The programmes shall cover basic training in order to qualify for a certain position and refresher training as needed.

Representative full-scope simulator facilities shall be used for the training of control room operators to such an extent that the hands-on-training of normal and emergency operating procedures is effective. The simulator shall be equipped with software to cover normal operation, anticipated operational occurrences, and a range of accident conditions.

For control room operators, initial and annual refresher training shall include training on a representative full-scope simulator. Annual refresher training shall include at least 5 days on the simulator.

3.18.1.3 REQUIREMENTS FOR VERIFICATION AND DEMONSTRATION OF DESIGN VALIDITY

Reference: IAEA SSR 2/1 Req. 32, 5.62

Verification and validation (including by simulators) of features relating to human factors must be included at appropriate stages to confirm that necessary actions by the operator have been identified and can be correctly performed.

3.18.1.4 REQUIREMENTS FOR THE IMPLEMENTATION AND OPERATING STAGE

The legislation area specified by assignment does not contain specific requirements for the implementation and operation stage. This shall be specified in more detail within the following stages of the licensing documentation.

3.18.2 PROPERTIES OF THE HUMAN FACTORS ENGINEERING DESIGN FOR THE NEEDS OF THE PRELIMINARY ASSESSMENT

This section describes the properties of the design for the needs of a preliminary assessment of the design concept. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant design. This section includes partial assessment as to whether the preliminary concept of the design segment in question complies with legislative requirements specified in Sections 3.18.1.1 to 3.18.1.4. The scope of this section includes identification and assessment of the general level of the requirements for the functions and principles of engineering psychology-ergonomics whereas particulars of the specific technological implementation shall be included in the design documentation of the selected supplier of the NPP, and the assessment shall be performed within the next stage of the safety documentation.

The design in the area of human factors engineering shall be designed so that meeting of required safety functions stipulated by binding legislation is secured. Legislative requirements derived from specified legislative supporting documents for processing of ISAR are summarised in Sections 3.18.1.1 to 3.18.1.4.

General principles of the design in the area of human factors engineering effective for all service stations assumed in the design solution are stated in Section "3.18.2.1 Planning and management of approach to human factors engineering in the preliminary design concept".

The design shall include design of structures, components, and systems relevant for the area of human factors engineering specified in Sections 3.18.2.1 to 3.18.2.4, whereas the particular method of their implementation shall be specified in the draft of the particular design solution of the nuclear power plant. The design solution selected for implementation does not have to include all systems, structures, and components specified in this Section; however, if the selected design utilizes them, they shall fulfil applicable requirements specified in this Section.

3.18.2.1 PLANNING AND MANAGEMENT OF APPROACH TO HUMAN FACTORS ENGINEERING IN THE PRELIMINARY DESIGN CONCEPT

The power plant design shall in a controlled way deal with aspects of human factors engineering. Respecting of functions and principles in this area shall secure balanced accordance between technical means intended for monitoring and management of technological systems of the process of electric power generation in a nuclear power plant and working conditions for power plant operating staff.

The systems relevant for the areas of human factors engineering shall be designed in such a way that they are fully in conformity with stipulated requirements, i.e. they secure support to operating staff when they perform their duties and tasks with the objective of limiting the human factor impact on nuclear safety.

For meeting the requirements of human factors engineering, the ETE 3,4 design shall use the technical infrastructure of inspection and control systems, i.e. systems, structures, and components and their equipment integrated within the instrumentation and control systems. The required design solution of control and management systems is specified in Section 3.7.2 Comprehensive preliminary assessment of the. design concept and is not the subject matter of Section 3.18.2.

Systematic approach to the design

The nuclear power plant design shall emphasize the area of man-machine interface already in the early stage of the design and further then throughout the whole project. In the course of the designing process, analyses shall be carried out, which shall ensure meeting of requirements for man-machine interface by means of balanced relation between people and power plant system interfaces.

The required design solution shall in its process deal with man-machine interface (MMI) in conformity with the following minimum requirements:

- MMI project team: The contractor shall create within its organization one integrated and cross-department project team, which shall have the overall responsibility for MMI projects (definitions, supervisions, review, approval)
- Task Analysis implemented on a functional basis: the contractor shall use this approach for division of HW and SW and also for operational processes, defining of displays and alarms, etc.
- Verification and validation: The V&V process shall include the use of models and simulators
- SW configuration management shall be established

The design shall specifically define elements that shall be integrated in the course of the designing process to secure systematic considerations in the area of human factor.

The contractor shall adopt such measures that the operating and maintenance personnel and the representatives of the construction and start-up personnel of the future plant operator may participate in the whole process of design development and implementation.

The contractor shall specify in the design assignation of each task to central or local management level. This decision shall take into account economic factors, legislation, working conditions and environmental conditions, ergonomic factors, overview of possibilities and skills, maintenance and inspection and also the degree of automation. The main requirement of the design is that the control room staff have continuous awareness of the power plant safe state in all operational states that may be used for control room operations.

Based on the carried out analysis, the contractor shall determine the adequate degree of automation for each required task. Automation degree analysis shall be implemented as a part of Task Analysis. The degree of automation and ergonomics in the design shall be selected in such a way as to minimize the probability of human error and that the consequences of such error (should it occur) are limited.

The required design solution shall secure that the designed working areas at work stations of the operating staff, their layout, access to requisite information and control means and presentation of information shall be designed with regard to the human factor (engineering psychology).

Reflection of results of functional analyses

The required design solution shall secure that in the course of the whole designing process (from the initial concept via the stages of initial and detailed design up to construction and testing) functional analyses shall be performed.

The objective of functional analyses is to determine functional objectives with regard to work force, technology and other sources and to provide a basis for determination as to how the required functions may be assigned and performed.

These analyses shall be implemented with the objective of securing the following:

- identification of all requisite functions and tasks in order to achieve the determined objectives and requirements,
- optimal division of tasks between the operating staff and automated systems,
- performance of tasks assigned to the operating staff in all operating conditions and optimal support to operating staff using requisite tools
- proposal of MMI design according to best knowledge in the ergonomics area

Task Analysis

The required design solution shall secure that the design process shall be based on task analysis and shall be implemented within the responsibility of one organization and one project team (see minimum requirements for design solution of MMI interface above). It shall be secured that the task analysis shall be implemented with the objective to specify detailed parts of all functions and all characteristics of each identified task. In the design, this process shall be applied to man-machine interface and shall not depend on the purpose of the interface (i.e. the part of the power plant, e.g. primary and secondary part), or the safety classification of the equipment or placement of MMI interface (e.g. control room and supplementary control room).

Measures shall be adopted in the design so that the ergonomic aspects of MMI are solved in the early stages of the project. The designer shall take into account the human factor in the course of the whole designing process of the control room and instrumentation and control systems.

Design principles for man-machine interface

Basic designing principles of the MMI design process shall be applied in the design.

The required MMI design solution shall respect the following main principles:

- optimum harmonization of tasks and tools within ergonomic conditions, e.g. within anthropometric, physiological and cognitive aspects,
- greatest possible restriction of the number of various MMIs,
- respecting of human qualities and characteristics and in particular in cases of accidents and process control the use of abilities of instrumentation and control systems for performance of operational and safety tasks within the highest possible scope without overloading the operator,
- operationally focused layout of work stations for management and control in the following configurations:
  - centralized permanently attended control room,
  - local control work station operated upon request,
  - supplementary control room attended upon request.

Besides these factors, the required design solution shall take into account the use of the most modern technology for instrumentation and control systems present another design basis for implementation of man-machine interface.

The design shall secure that the operating staff have the following information available:

- overview information on mutual links and operational sequences for the purposes of monitoring of processes and
- detailed information for process management

The process of MMI design is a translation of functional requirements and task requirements into the detail required for the MMI project, this is particularly the role of:

- alarms,
- display devices,
- control elements and
- task support means

which together create MMI. MMI design shall be the result of a process which takes into account functional requirements, task requirements, operational considerations (e.g. within the context of all tasks in which MMI shall be used) and personal safety and staff convenience.

The scope of the MMI project shall include the following:

- overall working environment
- working area layout at work stations (e.g. control room, supplementary control room, technical support centre)
- design of control panels and control boards

- design of layout of control and display devices
- detailed design of information and control interface such as the formats of graphic display, symbols, design communication and methods of entering inputs

Partial preliminary assessment

The preliminary concept of the design summarising the key requirements for systems, structures and components, and safety and technological functions of the instrumentation and control systems specified in Section 3.18.2.1 creates prerequisites for compliance with the legislation requirements of Decree No. 195/1999 Coll. [L. 266] Article 16(1)(2)(3), document IAEA SSR 2/1 [L. 252] Req. 32 (5.54, 5.55, 5.56, 5.57, 5.58, 5.59, 5.61, 5.62), 59, 60 (6.31) and Safety Guide SÚJB BN-JB-1.0 [L. 276] (83, 84, 95).

3.18.2.2 DESIGNING - CONTROL ROOMS AND SUPPORT CENTRES IN THE PRELIMINARY CONCEPT OF THE DESIGN

The nuclear power plant design shall include a control room, a supplementary control room and support work stations.

The requested design solution of habitability and environment of the operating and supplementary control rooms is stated in Section 3.6.2.4 and is not the subject matter of Section 3.18.2.2.

Control room

The required design solution of the power plant shall secure that the control room is placed in order to allow its easy use in all conditions. This design shall take into account external and internal impacts (see Section 3.3.3 Protection against the external hazards in the preliminary design concept and Section 3.3.5 Protection against internal hazards in the preliminary design concept).

The requested design solution of the control room shall ensure that instruments enabling safe operation of a nuclear unit in normal conditions shall be available in the control room and, at the same time, these instruments shall enable to switch the unit block to a safe state after occurrence of accident conditions or design extension conditions.

The design solution of workstations, their layout, a method of access to information and control instruments, and presentation of information shall be designed in accordance with principles of human factors engineering.

The process of design of the Man-Machine Interface (MMI) shall consider all required functions and tasks and shall include them in the detailed MMI design.

The Man-Machine Interface design shall be a result of a complex process, in which all required functions and tasks, operation aspects, and safety and comfort of workers will be taken into consideration.

Information and control means in the control room

The required design solution of the control room shall secure that control and information means are available at the control room necessary for operation of the power plant in the course of all operating states including start-up, maintenance and accident conditions. The control room shall also be used for controlling in the case of design extension conditions (but with the possibility of local interventions).

The information and control means in the project shall be designed in such a way that the operating staff may perform their tasks in the due time and manner.

The design of information and control means shall be focused in such a way as to provide stable and balanced assignment of tasks among operating staff and the control room, particularly for the whole scope of operational events and accidents (including partial loss of control and managing systems, loss of power supply, shutdown condition, etc.).

The design solution shall secure that information and control elements shall be logically and functionally arranged in the same way as the feedback elements for equipment and/or process status (same format or the same elements on the panel).

The design solution shall secure that the operator may at any time become sufficiently familiar with the current state of the power plant by means of information and control elements (overview diagrams on conventional panels, displays on the computer control means).

The project shall pay special attention to manual control of basic protection channels (reactor and turbine shutdown, emergency injection, etc.). Unless substantiated by any other technical requirements, the design shall for psychological reasons contain elements of manual control besides computer soft controls.

The control room shall include a large overview display showing a diagram of the power plant. This display shall permanently show information about the main parameters of the power plant and its status. The information on this overview display shall be legible from every work station of the control room.

Supplementary control room

The design shall also include a supplementary control room that shall be utilized when the control room is not accessible. With regard to events that require operation of the supplementary control room, the designs shall draft its functions and placement in such a way that the control room is accessible by safe access routes. The required power plant design shall secure that simultaneous loss of the ability to perform safety functions from the operation and supplementary control room does not occur due to external or internal influences (see Section 3.3.3 Protection against the external hazards in the preliminary design concept and Section 3.3.5 Protection against internal hazards in the preliminary design concept).

The design shall secure that simultaneous controlling from the control room and the supplementary control room shall not be possible. This requirement shall be secured by means of technical means or administrative measures. At the same time measures shall be taken to prevent impact of false signals from the supplementary control room in the case of controlling from the control room. The supplementary control room design shall include means to restrict unauthorized use of the supplementary control room. At the same time, access to the supplementary control room shall be indicated in the control room.

Supplementary control room design shall be implemented in conformity with human factor (human factors engineering) principles and respecting human qualities (characteristics) in emergency conditions.

MMI design solution in the supplementary control room shall be designed in a similar way as the MMI used in the control room.

Supporting facilities

The required power plant design shall include work stations intended for dealing with accident states of the power plant and for support activities to operating staff in the case of extraordinary events.

The required power plant design shall include complementation of the existing emergency control room (emergency commission and technical support centre).

The project for these support work stations shall secure the following:

- premises and equipment for activities of support work teams specified by on-site emergency plan of the NPP (meetings, administrative activities, requisite documentation, etc.)
- information and equipment necessary for supporting of accident plans in the case of design extension conditions (DEC) including severe accidents (SA) - e.g. MMI station with access to technological data (only information, not controls)
- information and equipment necessary for support of operating staff in the case of design basis accidents (DBA) - e.g. MMI stations with access to technological data (only information, not controls)
- information and equipment necessary for determination of unit condition (in the normal operation even in the case of extraordinary event)
- information and equipment necessary for evaluation of radiation and fire safety in the power plant
- information and equipment necessary for identification and quantification of radioactive and ionisation releases and doses
- information required for communication with state authorities and their accident centres (technological, radiation and meteorological data)
- equipment for visual inspection (camera system) of equipment of main technological systems, control rooms and supplementary control rooms
- communication means (equipment for visual and verbal communication) for communication with control room, supplementary control room and means for other external communication in conformity with the requirements of the on-site emergency plan of the NPP

Emergency control centre

The required design solution of the power plant shall include an emergency control centre - premises and equipment for activities of accident staff of the power plant.

The design shall ensure that the accident centre is habitable even in the cases of design basis accidents (DBA) and design extension conditions (DEC).

Technical support centre

The required power plant design shall include a technical support centre. This centre shall be available in emergency situations for the technical support team (consulting and advisory staff) and its objective shall be to perform functions and conditions defined by the on-site emergency plan of the NPP approved by SÚJB.

The design shall ensure that the technical support centre is habitable even in the cases of design basis accidents (DBA) and design extension conditions (DEC).

Staff, training equipment and programmes

Staff, training and programmes

The design shall include training of operating personnel at least in the scope of the staff of the control room, technical support centre, shift engineer, nuclear physics specialists, system engineers, quality assurance workers, radiation safety workers, operating workers, chemistry workers, maintenance staff (construction, engineering, instrumentation and control systems) for the nuclear and conventional part of the power plant and all workers for power plant start-up.

The contractor shall provide qualification requirements and description of working activities for the power plant staff and based on that the contractor specifies training requirements for every position. The design shall specify a list of requirements and objectives for each type of training in the given design stage.

The design shall ensure that training is performed in relation to the particular stage of the design.

Training shall be divided into three stages:

- Stage 1 – the contractor shall train the staff intended for cooperation in the course of designing and construction stage and trainers for the purposes of further training on the customer's side
- Stage 2 – the contractor shall train selected staff members for start-up, operation and maintenance
- Stage 3 – training within the power plant operator's responsibility – training of other staff members, refresher training, etc.

Full scope simulator

The required power plant design shall include a full-scope training simulator for ETE 3,4. The full-scope simulator shall be intended for training and qualification improvement of operating staff before commissioning the unit and for regular staff training.

The design of the training simulator shall include a replica of the control room and the supplementary control room.

The proposed design of the training simulator shall include full and exact simulation of all systems characteristics of ETE 3,4 in operating conditions and design basis accidents and shall include the scenario of design extension conditions (DEC). The system model shall contain all interfaces and links to other simulated systems and shall provide a realistic integrated simulation of operation of ETE 3,4.

The design of the simulator shall include simulation of defects and breakdowns in all power plant processes and its instrumentation in such a way that the operator fully learns mutual synergies that may occur during an accident in the reference unit.

The contractor shall secure readiness of the training simulator for acceptance tests for implementation of training (i.e. readiness of the equipment and SW) at least 364 days before starting fuel loading.

Partial preliminary assessment

The preliminary concept of the design summarizing the key requirements for systems, structures, and components, and safety and technological functions of the instrumentation and control systems specified in Section 3.18.2.2 Designing - control rooms and support centres in the preliminary concept of the design creates prerequisites for compliance with all the legislative requirements specified in the partial preliminary assessment of Section 3.18.2.2 and other specific requirements of Decree No. 195/1999 Coll. [L. 266], Article 20 (1)(2)(3), document IAEA SSR 2/1 [L. 252] Req. 32 (5.53, 5.60, 5.61), 65 (6.39, 6.40), 66 (6.41), 67 (6.42) and Safety Guide SÚJB BN-JB-1.0 [L. 276] (93, 94, 95, 96, 97).

3.18.2.3 VERIFICATION AND PROVING OF VALIDITY OF THE PRELIMINARY DESIGN CONCEPT

The design shall include measures in order that activities are incorporated for verification and demonstration of design validity in the designing process after each designing stage. The objective of these activities shall be to ascertain that the requirements of the given stage are duly fulfilled before starting another stage.

The activities for verification and demonstration of validity of the design shall be a part of the overall quality assurance programme that shall secure proper incorporation of functional and other requirements (technical, ergonomical and human factor) in the design of the instrumentation and control system and the control room.

A full-scope simulator shall be used in the designing process for verification and demonstration of design validity.

Partial preliminary assessment

The preliminary design concept summarizing the most important requirements for verification and proving of the validity of design of systems, structures, and components, safety and technological functions relevant in the area of human factors engineering stated in Section 3.18.2.3 creates prerequisites for fulfilment of all legislation requirements stated in the partial preliminary assessment of Section 3.18.2.1 and further specific requirements of document IAEA SSR 2/1 [L. 252] Req. 32 (5.62).

3.18.2.4 IMPLEMENTATION AND OPERATION STAGE IN THE PRELIMINARY DESIGN CONCEPT

Preliminary assessment of the design concept in the area of the implementation stage and operation of structures, components and systems relevant for the area of human factors engineering is not implemented as the delimited scope of binding legislation does not stipulate any requirements in the assessed area.

3.18.3 SUMMARY PRELIMINARY ASSESSMENT OF HUMAN FACTORS ENGINEERING DESIGN CONCEPT

The ETE 3,4 design shall meet the safety requirements specified in Section 3.3.1.1 Basic legislative requirements for provision of safety as well as the requirements specified in Sections 3.18.1.1 to 3.18.1.4, which are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and related implementing decrees as regards nuclear safety, radiation protection and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252] and WENRA [L. 27] and [L. 270] documents.

In order to meet the above requirements, the ETE 3,4 design shall thoroughly take into account functions and principles of engineering psychology-ergonomics based particularly on functional analysis and division of tasks between management systems and power plant operators. To meet the aforementioned requirements, the design of ETE 3,4 shall use the technical infrastructure of instrumentation and control systems, i.e. systems, structures, and components, and their facilities integrated within the instrumentation and control systems that will ensure (with adequate reliability and resistance) qualitatively and quantitatively adequate monitoring, control, and protective functions of technical systems (especially mechanical-technical, construction, and electrical systems), including systems important to nuclear safety. The ETE 3,4 design shall secure comprehensive application of functions and principles of engineering psychology-ergonomics and their integration in requirements for instrumentation and control systems in conformity with stipulated requirements for performance of specified safety functions.

The principles of the design solution described in Section 3.18.2 Design characteristics for the needs of the preliminary assessment were set up based on the requirements submitted by the applicant for licence applied on the potential suppliers of the nuclear installation within the tender, and they make up the concept of the design solution for this part of the design. The partial assessments performed proved that the expected design creates prerequisites for compliance with the relevant requirements for systems, structures, and components and safety and technological functions specified by Decree No. 195/1999 Coll. [L. 266], SÚJB BN-JB-1.0 Safety Guide [L. 276], document IAEA SSR 2/1 [L. 252], and document WENRA [L. 27]. Systems, structures, and components relevant for the area of human factors engineering conform to the implemented preliminary assessment of the design concept.

The particular method of technical implementation of individual requirements for principles of engineering psychology-ergonomics specified in Section 3.18.2 shall be specified in detail only in the design documentation of the selected nuclear power plant contractor.

3.19 PROBABILITY ANALYSES AND ASSESSMENT OF DESIGN EXTENSION CONDITIONS

Section 3.19.1 briefly states the scope and principles for drawing up probability safety assessment (PSA), which shall be presented to SÚJB together with the requirement for construction of a new nuclear power source. The PSAR attached to this application shall include the total results of probabilistic analysis, whereas the actual PSA study shall be available for SÚJB assessment as a separate document. The results shall also include demonstration of project conformity with the stipulated probability safety objectives.

In conformity with the current international safety requirements, the adequacy of the design shall be demonstrated even for very improbable accident conditions including the scale of design extension conditions with severe accidents. Section 3.19.2 briefly states the scope and principles for implementation of analyses of groups of events for any of the designs that may be considered in the current tender. The objective of the section is to formulate the design's general characteristics for the purposes of partial preliminary assessment. The actual analyses drawn up in conformity with the principles stated further shall be subsequently a part of the PSAR to be submitted to SÚJB together with the requirement for permission to build a new nuclear source.

Using the sum of requirements for safety analyses defined within Section 3.19.2, Section 3.19.3 states the preliminary assessment of design concept required by law.

3.19.1 BASIC LEGISLATION REQUIREMENTS FOR PROBABILISTIC ANALYSES AND ASSESSMENT OF SEVERE ACCIDENTS

3.19.1.1 PROBABILISTIC ANALYSES

Reference: Safety Guide SÚJB BN-JB-1.0 (27), (30), WENRA Issue O 1.1, 2.1

For the new nuclear plant design, a specific PSA shall be developed of probability of occurrence of design extension conditions occurrence expressed as frequency of occurrence of severe damage to the fuel system (PSA level 1) and probability analysis of occurrence of radiation accident focused on assessment of response to PSA level 1 accident scenarios stating the frequency of occurrence of early and large releases of radioactive substances outside the containment including their quality and quantity characteristics (PSA level 2). These analyses shall be implemented, documented and maintained in conformity with requirements for probability analyses stated by the State Office for Nuclear Safety.

Reference: Safety Guide SÚJB BN-JB-1.0 (27), (30), IAEA SSR 2/1 5.76, WENRA Issue O 3.3

The objective of the probabilistic safety analysis shall be determination of all significant factors which contribute to the radiation risk resulting from the nuclear power plant, particularly the following:

- confirmation that barriers and in-depth protection levels satisfactorily perform their functions
- confirmation that the design is balanced, i.e. that no individual characterisation or no postulated initiating event causes an inadequately large or significantly unclear contribution to the overall risk
- confirmation that there is sufficient prevention against impacts of small parameter deviations of the nuclear installation which might cause significant changes of its operating conditions
- confirmation that within a practically achievable scope the measures at individual in-depth protection levels are independent
- evaluation of acquired analysis results by comparing with probabilistic safety objectives stipulated by the State Office for Nuclear Safety

Reference: Safety Guide SÚJB BN-JB-1.0 (7), (23), (36), (47), IAEA GS-R-4 4.55, 4.56, WENRA Issue O 3.1, 3.2, 3.4, 3.5, 3.6, 4.1, 4.2, 4.3

Probabilistic modelling and assessment is to be performed in such a way as to provide sufficiently detailed information on reliability of the nuclear power plant resource, mutual links of its individual parts and possible weak spots of the design, achieved in-depth protection and possible risks. Probabilistic analyses shall be processed in the scope that shall allow their use

- as the sources of information for decision-making within safety management
- for determining operability requirements and safety classification of equipment
- for determining requirements for tests of equipment and determining the permissible period for decommissioning the equipment
- for setting up a list of postulated initiating events that may have a significant impact on nuclear installation safety including those that may be caused by external or internal influences caused by natural occurrences or human activities or combination of both
- for selecting representative scenarios of accidents, particularly in the category of design extension conditions including severe accidents
- for identifying requisite needs and evaluation of adequacy of power plant equipment modifications and operating regulations including measures to deal with severe accidents and to assess severity of operational events
- for preparing supporting materials and development and validation of safety significant training programmes of the operator, including simulator training of the control room staff. In all cases of utilization of probabilistic analyses their adequacy and possible limitations for the given purpose shall be duly considered

Reference: Safety Guide SÚJB BN-JB-1.0 (30), (139), WENRA Issue O 1.2, 1.3, 1.4, 1.5, 2.2, S 3.4

PSA shall be drawn up using methodology in conformity with the requirements of the State Office for Nuclear Safety considering the currently available international experience. Level 1 PSA shall contain sensitivity analysis and uncertainty analysis. Level 2 PSA shall include min. sensitivity analysis and uncertainty analysis where appropriate. PSA shall include all operating conditions of the nuclear installation and all substantial initiating events including impact of internal fires, floods, adverse climatic conditions, seismic events and human interventions. Significant internal dependencies shall be included (i.e. functional and spatial links and other failures due to a common reason). Analysis of human activity reliability shall be implemented considering factors that may affect the activities of operating staff in all individual operational states.

3.19.1.2 ASSESSMENT OF DESIGN EXTENSION CONDITIONS

Reference: Safety Guide SÚJB BN-JB-1.0 (27), (38), GS-R-4 4.50, WENRA App. F.1.1

In the PSAR, nuclear safety shall be assessed even for design extension conditions including severe accidents with the objective to demonstrate efficiency of precautionary and mitigating measures for preventing of exposure of persons and impacts on the environment even in the case of events with very low occurrence frequency but with potentially substantial consequences. These events may be formed as a combination of initiating events and failures not included among design basis accidents (e.g. multiple breakdowns, total loss of safety systems, subsequent loss of safety systems in the case of long-term development of an accident). The course and radiation consequences of severe accidents not having the character of practically eliminated conditions shall be assessed:

- to identify practically feasible measures for prevention of formation and development of accidents and for management and mitigation of their consequences
- as a basis for drawing up manuals designed for managing of accidents and for staff training
- as a basis for drawing up plans for protecting the staff and the population, and implementing mitigating measures to reduce the impact of radioactive releases threatening the staff, population and the environment

The analyses shall be implemented in conformity with the current requirements of the State Office for Nuclear Safety. Preliminary requirements for analyses of design extension conditions are stated in the following text.

Reference: IAEA SSR 2/1 5.27

Managing of design extension conditions might require additional safety features for design extension conditions, or extension of the capability of safety systems to maintain the integrity of the containment. These safety features shall be such as to ensure the capability for managing accident conditions in which there is a significant amount of radioactive material in the containment (including radioactive material resulting from severe degradation of the reactor core). The plant shall be designed so that it can be brought into a controlled state and the containment function can be maintained, with the result that significant radioactive releases would be practically eliminated.

Reference: Safety Guide SÚJB BN-JB-1.0 (37), (39), IAEA SSR 2/1 5.31, GS-R-4 4.57, WENRA App. F 2.2

Acceptance criteria and processes for determination of acceptability of safety analyses results shall be determined for design extension conditions analyses. The acceptance criteria for analyses of these conditions may be stipulated in a less conservative manner than with the design basis accidents. The analyses shall prove meeting of these criteria so that

- severe accidents that may result in early or substantial releases of radioactive substances are practically eliminated
- in the case of severe accidents that do not have the nature of practically excluded conditions and whose radiation consequences may be substantial, it is ensured that only limited protection measures shall be necessary for the population, i.e. it shall not be necessary to evacuate population from the immediate vicinity of the power plant; in the extreme case, only time limited sheltering of the population shall be necessary and no long term restrictions shall be required regarding the nuclear installation and so that there shall be sufficient time for application of these measures.

Reference: Safety Guide SÚJB BN-JB-1.0 (36), WENRA App. F 2.1

Using the combination of deterministic and probabilistic methods and engineer's opinion the selection of the most significant design extension conditions shall be made, they shall be subject to safety analyses and the events for which it is necessary to and reasonably feasible to establish in the nuclear installation design the corresponding precautionary or mitigating technical and organizational measures shall be determined. The selection of types of events within design extension conditions that need to be analysed to verify design safety unless they are a part of a set of abnormal events and design basis accidents shall include at least the following:

- abnormal operation events with postulated failure of the safety system of rapid reactor shutdown
- complete long-term loss of internal and external power supply
- complete long-term loss of feed water supply
- accident with loss of coolant with simultaneous loss of emergency cooling of reactor active zone (high-pressure or low-pressure cooling system AZ)
- uncontrolled drop of level during operation with decreased level of coolant in the reactor (mid-loop operation) or during refuelling
- complete loss of cooling of components (intermediate closed coolant circuits)
- loss of system of reactor residual heat removal
- loss of cooling of the spent fuel storage pool
- loss of final heat sink
- uncontrolled concentration of boric acid in reactor
- multiple disruption of steam generator tubes

- loss of requisite safety systems in the case of their long-term use after an initiating event

Within the groups above, representative scenarios shall be selected of the development of a severe accident of the given reactor with the most severe radiation consequences or the most probable scenarios to be assessed starting with the loss of cooling of the active zone or storage pool and development of fuel melting in order to ascertain the limit response of the nuclear installation and prove efficiency of technical and administrative measures in these accident conditions.

Reference: Decree No. 195/1999 Coll., Section 4 (3), IAEA GS-R-4 4.60

The quality and suitability of computational programs used for the analyses important for nuclear safety shall be verified within sufficient scope within available means in a similar way as required for analyses of design basis accidents.

Reference: IAEA SSR 2/1 5.29, 5.30

The analysis must confirm sufficient reliability and efficiency of measures for preventing or dealing with design extension conditions which must be (to a practically achievable degree) independent of measures used for management of design basis accidents. It shall be verified whether severe accidents may be managed only by using specific systems designed for management of severe accidents. However, in general, it is possible to consider using of any operable power plant systems for managing of design extension conditions even beyond the scope of their normal function and operational parameters if the ability of the systems to execute the required activities and ability to survive even in the conditions of these accidents is demonstrated.

Reference: IAEA SSR 2/1 5.75 (6)

Deterministic analyses must prove particularly that using the automatic function of safety systems in combination with expected operator interventions, it is possible to manage design extension conditions.

Reference: Safety Guide SÚJB BN-JB-1.0 (29), (37), (40), IAEA GS-R-4 4.48, 4.54

The requirement for sufficient safety margins in deterministic analyses also concerns the design extension conditions including severe accidents with which it is otherwise possible to apply a realistic approach to analyses. Potentially larger uncertainty of results of analyses due to uncertainty in models, initial and limit conditions shall be respected by using adequate reserves when using results for determining the time development of the accident and the scope of their consequences taking into account uncertainties of input data and used analysis methods. This procedure secures with a great degree of credibility that the radiation risk level for staff and population shall be admissibly low.

3.19.2 PROBABILISTIC ANALYSES AND ASSESSMENT OF SEVERE ACCIDENTS FOR THE NEEDS OF PRELIMINARY ASSESSMENT

This section includes the design characteristics in terms of compliance with the basic requirements for probabilistic analyses and assessment of severe accidents for the purposes of the preliminary design concept assessment. The information for the specification of the design characteristics was based on the technical part of the BIS stipulating the requirements for safety and technical design of the future power plant design. This section includes partial assessment as to whether the preliminary concept of the design segment in question complies with the requirements specified in Sections 3.19.1.1 to 3.19.1.2. The subject matter is the assessment of compliance with general level of requirements for implementation of probabilistic analysis and assessment of severe accidents whereas the particular method of implementation of these analyses shall be specified in the nuclear power plant design and evaluated in the following stage of safety documentation.

The basic requirements for the method of implementation of probabilistic analysis and assessment of severe accidents is based on the general requirement for application of in-depth protection (in-depth protection requirements are specified in Section 3.3.1.1.3 and the requirements for the process of implementation of in-depth protection are specified in Section 3.3.1.1.4).

The method of implementation of probabilistic analysis and evaluation of severe accidents in the PSAR shall meet the minimum requirements stipulated in binding legislation.

3.19.2.1 PROBABILISTIC ANALYSES IN THE PRELIMINARY DESIGN CONCEPT

In order to verify the following probabilistic objectives, a probabilistic analysis shall be implemented of level 1 safety (leading to ascertainment of frequency of severe damage to fuel system) and level 2 (leading to ascertainment of frequency and scope of radioactive releases):

- summary frequency of severe damage to fuel system shall be below 10-5/year
- summary frequency of radioactive release exceeding the radiation acceptance criteria for design extension conditions (see chapter 4) shall be under 10-6/year
- summary frequency of early (early failure of containment system) or large radioactive releases (one order higher than the radiation acceptance criteria for design extension conditions) shall be sufficiently below 10-6/year
- the frequency of any individual scenario of accident development with radioactive release exceeding the acceptance radiation criterion for design extension conditions shall be 10-7/year

Despite the fact that the analysis is to be focused primarily on determination of frequency of accidents with severe damage to the fuel system and associated releases, it shall also include releases from other sources of radioactive materials in the power plant (e.g. storing and transport of fuel including the events initiated in spent fuel storage pool, processing of radioactive waste, etc.).

Moreover this analysis shall be used:

- for verification of sufficient reliability of equipment intended for managing of severe accidents,
- as a supplement for the deterministic safety analysis when assessing frequency of initiating events and their combinations
- for identification of accidents with multiple breakdowns and the draft of measures for managing these defects

- for support of determination of operational limits and conditions, emergency procedures and instructions for managing of design extension conditions
- for achieving of a balanced design

PSA shall be also used for assessment of reliability of redundancies and assessment of possibility of breakdowns with a common cause between redundant branches.

In order to obtain real probability values, PSA shall consider equipment important and not important to safety. PSA shall be used for determination of the most important non-safety equipment in terms of decreasing of frequency of accidents and for the equipment determined in this way (particularly for projects with fully passive safety systems) measures shall be determined to increase their reliability.

Reliability of the considered equipment shall be substantiated by reliability studies. Reliability data from operation of current power plants and other relevant sources shall be used and analysed so that they may be verified.

PSA shall take into account operator's error (both the diagnostic error and error when implementing activity according to operating and accident specifications).

Contributions of individual events to the frequency of severe damage of fuel system or early release shall be compared in order to ascertain that there is no dominant risk and a balanced project has been achieved.

PSA shall be managed in such a way so that in operation it remains the so-called live analysis, which reflects the current state of the power plant and may be used for:

- assessment of operational limits and conditions
- assessment of operating processes
- assessment of accident processes and instructions for design extension conditions
- development and optimisation of accident preparedness programmes
- maintenance optimization
- prioritisation of checks and tests
- assessment of operational experience
- assessment of design changes
- support to intervening staff in abnormal operation and accidents
- support to simulator training

Initiating events

The analysis shall include all initiating events including power states, low-power states and states with shut down reactor. All events stated in the list in Section 3.15.2 shall be nominally included. Internal and external forces shall be also considered according to the scope in the list in Sections 3.3.3 to 3.3.5 (excluding sabotage). Some external influences might be excluded from detailed evaluation based on low frequency (below 10-7/year) or based on verified engineering criteria proving resistance of relevant physical barriers against damage.

Moreover, a systematic and full-range analysis shall be implemented to identify initiation events including, e.g.:

- engineering assessment (e.g. FMEA) of operating and safety systems
- evaluation of operating experience from the current generation of power plants
- research of PSA analyses of power plants of similar design

Assessment of internal influences

Internal influences shall be within the scope according to the list in Sections 3.3.3 to 3.3.5 considered in PSA for both power states and shut down states. These internal influences are fires, explosions, releases of gases and liquids including flooding, release of steam or harmful substances, failure of pressure equipment, civil structures, failure of rotary machinery including the turbine, falls and impacts of bodies and electromagnetic interference.

Frequency of severe damage to the fuel system shall be calculated for all internal hazards that may not be excluded based on probability criterion - frequency of exceeding of radiation acceptance criteria for design extension conditions caused by a group of internal events is under 10-7/year.

External events

External events according to Section 3.3.3 to 3.3.5, excluding sabotage, shall be analysed. These events include natural events (earthquake, floods, extreme temperatures, extreme wind, extreme precipitation, droughts or lightning) and events caused by human activities (plane crash, industrial activity, electromagnetic interference).

Analysis shall also be applied to possible combinations of external events, e.g. strong wind and external power supply outage as a result of its effect, snow fall and icing and similar occurrences which may occur simultaneously.

Seismic PSA

When assessing earthquake, seismic margin assessment shall be implemented.

Moreover, the most significant initiating events caused by earthquake shall be assessed as specified in seismic margin assessment. Assessment shall be realized using modified PSA of internal events level 1 and 2. The objective is to provide a reasonable estimate of seismic risk based on SMA results and demonstrate the impact on PSA probability objectives.

Human reliability analysis

PSA shall also include human reliability analysis.

The analysis shall include at least the following:

- systematic identification of operators' activities important in terms of risk (using qualitative and quantitative analysis)
- quantification of operators' activities identified as important in terms of risk
- sensitivity analysis (including identification of necessary operators' activities and analysis of necessity of possible interventions within 30 minutes of start of the event)

- basis for completion of the analysis during development of operating and accident procedures

Analyses of uncertainties, sensitivity and importance

Besides enumeration of point values, analysis of uncertainties, sensitivity and importance shall be performed.

The following areas of uncertainties and sensitivity shall be identified:

- Quantitative analysis of uncertainties for basic events in PSA model. The systematic quantitative analysis shall reflect uncertainties in numbers of frequency of initiating events, reliability data of components and operators' activities
- For further sources of uncertainties particularly model assumptions, e.g. success criteria, human error, grouping of failures with the same cause, a sensitivity analysis shall be performed. The areas are as follows:
    - assumptions of models of dominant accident sequences
    - uncertainties in models based on engineering judgement
- Sensitivity analysis by change of input data of initiating events (frequency) for which there is a great uncertainty of determination of frequency

Importance analysis shall provide the way of interpretation of PSA results. It shall be calculated for events and groups of events.

Conclusions and documentation

Results and insights obtained in PSA shall be described so as to allow assessment of the contribution to limitation of the risk of various safety measures and prove meeting of probability objectives. It shall contain discussion of changes in the design implemented based on PSA.

Detailed technical documentation shall provide sufficient data for possible reconstruction of PSA results. All intermediate stages, sub-analyses, calculations, assumptions, etc., that have not been published within the reports at higher levels shall be kept in this documentation in the form of reports, comments, input data, models and outputs.

PSA results shall be set up and presented so as to clearly show enumeration of the risk, aspects of the design and operation of the power plant that are most important in terms of the risk (for its formation and its limitation as well) and consequences of important sources of uncertainties. The presentation shall be complemented with charts and discussion over the most important PSA results for better understanding of insights obtained by means of PSA.

3.19.2.2 ASSESSMENT OF DESIGN EXTENSION CONDITIONS IN THE DESIGN PRELIMINARY CONCEPT

Safety analysis of design extension conditions shall be implemented with the objective to prove meeting of the following probability criteria:

- summary frequency of radioactive release exceeding the radiation acceptance criterion for design extension conditions (see chapter 4) shall be under 10-6/year
- summary frequency of early (early failure of containment system) or large radioactive releases (releases by one order higher than the radiation acceptance criteria for design extension conditions) shall be well enough below 10-6/year
- the frequency of any individual scenario of accident development with radioactive release exceeding the acceptance radiation criterion for design extension conditions shall be 10-7/year

It shall be demonstrated that for all considered design extension conditions that may not be excluded, the power plant shall contain effective systems for transfer into and keeping in a safe state.

Safety analyses shall be performed using the best approximation methods. All assumptions, parameters and calculation codes shall be based on acknowledged databases of experimental data or sufficiently substantiated contractor's databases. Securing of sufficient safety margins shall be achieved by using the results for determination of time development of the accident and the scope of their consequences.

In order to cope with design extension conditions it is possible to consider all equipment necessary for safety (i.e. not only safety systems) in the case that it is possible to assume preserving of their operability in the given conditions and after 72 hours even the equipment not important to safety. However, in analyses of severe accidents it shall be demonstrated in what conditions these accidents may be managed while complying with the stipulated criteria by using only specific systems intended for managing of severe accidents.

It shall not always be necessary to consider the criterion of simple breakdown especially if the justification of not using the criterion of a simple breakdown using probabilistic criteria proves to be correct. When determining the necessity of considering of the criterion of a simple breakdown, the possibility of repair during the accident and accessibility of the given areas shall be taken into account. It shall not be necessary to carry out full qualification of equipment to the design extension conditions. However, the so-called survivability of the equipment shall be demonstrated (see Section 3.3.11.2).

When implementing the assessment, the intervention of control room staff shall not be considered for 30 minutes and outside the control room for 60 minutes from occurrence of the accident.

Within design extension conditions, two types of events shall be analysed:

- multiple failures of systems without severe damage to the fuel system going beyond the events considered in design basis accidents
- severe accidents for preventing of early and late failure of containment system and limitation of radioactive releases

A selection of considered events shall be implemented for each of the stated categories using probabilistic methods, international customs or major uncertainties when determining occurrence frequency.

The category of multiple failures of systems shall include at least the following:

- anticipated transient without scram (ATWS)
- complete outage of power supply (station black-out)
- rupture of main steam line with subsequent rupturing of steam generator tubes
- accidents of containment bypass
- common cause failures (e.g. programmable components)

Further, the power plant's response to events described in document WENRA [L. 27] in Appendix F shall be assessed as described in Section 3.19.1.2.

For the case of ATWS, it shall be demonstrated that there is a suitable combination of internal (inherent) properties of the design and diverse means which sufficiently limit the frequency of such events and if they occur, they shall limit damage to the fuel and secure preserving of integrity of the pressure boundary of the primary circuit.

For the case of defects of the containment bypass system, it shall be demonstrated that the containment design is resistant enough, i.e. the frequency of such accidents is sufficiently low.

The measures to decrease the frequency of containment by-pass shall be as follows:

- sufficient margin of the strength of systems that may be connected to the primary circuit
- minimization of the number of bushings and passages through the containment system
- sufficient reliability and redundancy of valves on the piping connected to the primary circuit and going outside the containment system
- reliable safety equipment to reduce release in the case of steam generator tube rupture to prevent lifting or by-pass valve of the damaged steam generator and to facilitate operator's activities in these events

In the case of rupture of the main steam line with subsequent rupture of tubes in the steam generator, it shall be demonstrated that these accidents participate only minimally in the overall frequency of radioactive release exceeding the radiation acceptance criteria for design extension conditions. Determination of the number of tubes and associated assumptions of analyses shall take place based on realistic postulated mechanisms, assessment of best approximation and probability analyses.

In the category of severe accidents representative severe accidents shall be considered. These shall be selected based on probability criteria. In general, these shall be severe accidents contributing most to the overall frequency of severe damage to the fuel system. At the same time, severe accidents with lower frequency but much higher consequences shall be considered.

In the case of severe accidents it shall be demonstrated that there are effective measures in the design for preventing of severe accidents that result in failure of containment system particularly then severe accidents resulting in early failure of containment system followed by the following events:

- hydrogen explosions
- reactor pressure vessel failure due to high pressure
- steam explosion threatening integrity of the containment system
- accidents with great input of reactivity (including heterogeneous thinning of coolant in the primary circuit)

Further it shall be proved for the severe accidents that the containment system contains efficient measures for limiting of consequences of severe damage to the fuel system particularly in terms of:

- intercepting and cooling of melt
- interaction between the melt and concrete
- limitation of releases even when considering loading caused by oxidation of fuel coverage, hydrogen combustion and other strains caused by severe accidents
- extension of time when no operator intervention is necessary
-

3.19.3 COMPREHENSIVE PRELIMINARY ASSESSMENT OF THE DESIGN CONCEPT

Probability analyses and assessment of severe accidents shall meet the requirements that are based on the requirements stipulated by Act No. 18/1997 Coll. [L. 2] and its implementing decrees regarding nuclear safety, radiation protection, and emergency preparedness, as well as on the requirements specified in IAEA SSR 2/1 [L. 252], IAEA GS-R-4 [L. 278] and WENRA [L. 27] and [L. 270].

The assessment confirms that the assumed way of implementation of probability analyses and evaluation of severe accidents creates prerequisites for meeting relevant requirements specified by Decree No. 195/1999 Coll. [L. 266], Safety Guide SÚJB BN-JB-1.0 [L. 276], document IAEA SSR 2/1 [L. 252], document IAEA GS-R-4 [L. 278] and document WENRA [L. 27].

Specification of requirements for implementation of the probability analysis and evaluation of severe accidents characterized in Section "3.19.2 Probabilistic analyses and assessment of severe accidents for the needs of preliminary assessment was created based on requirements of the licence applicant applied to potential contractors of nuclear installation within the tender and creates the design concept of this particular design segment.

The particular method of implementation of probability analyses and evaluation of severe accidents shall be specified in detail in the following stage of safety documentation.

4 PRELIMINARY ASSESSMENT OF IMPACT OF OPERATION OF PROPOSED INSTALLATION ON PERSONNEL, THE PUBLIC AND THE ENVIRONMENT

4.1 PRELIMINARY RADIOLOGICAL SAFETY OBJECTIVES FOR NUCLEAR POWER PLANT DEFINED STATES

4.1.1 DESIGN DEFINED STATES OF NUCLEAR INSTALLATION

Preliminary assessment of impact of operation of the designed installation on personnel, the public and the environment is in the ISAR based on assumptions resulting from the BIS for maximum admissible radiation consequences of the design based defined states of the nuclear installation.

The requirements for safety parameters of new units are based on Czech legislation harmonized with EU legislation, statement of SÚJB regarding the Opinion on documentation of assessment of effects on the environment according to Act No. 100/2001 Coll. [L. 255] "New nuclear source in Temelín plant including power output to Kočín substation" and the document European Utility Requirements ("EUR") [L. 264]

THE APPROACH TO NUCLEAR SAFETY AND RADIATION PROTECTION INCLUDING SAFETY OBJECTIVES IS SUBJECT TO INCESSANT DEVELOPMENT DIRECTED TO HARMONIZATION OF DOCUMENTS OF CONCERNED INSTITUTIONS and organizations (Western European Nuclear Regulators' Association WENRA, WENRA RHWG – Reactor Harmonization Working Group, European Atomic Forum ENIISS – Initiative, IAEA, ICRP) and the results of this development also projected in EU directives and Czech legislation shall be accepted in further stages of the design and corresponding stages of safety documentation.

In ISAR, four basic possible states of the power plant are defined and we specify the safety objectives assigned to them:

Normal operation

Normal operation represents all states and operations of planned operation while observing operational limits and conditions of safe operation. NORMAL OPERATION INCLUDES (BESIDES POWER OPERATION) FREQUENT OR REGULAR SITUATIONS ASSOCIATED WITH PLANNED CHANGES OF POWER PLANT POWER OUTPUT.

Normal operation includes:

- operation in stable condition and shutdown,
- operation with admissible deviations within operational limits and conditions of safe operation and
- operation transient processes

It applies to normal and transient processes that all acceptance criteria are observed with sufficient margin forming the span between the actual value of the monitored parameter and its admissible limit value requiring intervention.

Abnormal operation

Abnormal operation means unplanned states, operations and events whose occurrence may be expected during operation of a nuclear installation (scramming, sudden loss of load, turbine outage, power supply outage, loss of main circulating pump, etc.). Abnormal operation may in the worst case result in reactor scramming whereas the power plant is able after termination of this state or removal of causes and consequences to return to normal operation (provided there is no damage to the fuel system, disruption of fuel elements or disruption of integrity of primary circuit).

The frequency of these events is estimated up to the level of ca 10^{-2} per year.

In terms of radiation protection, it is requested for operating states that exposure of a representative person and the environment is limited by the process of optimisation by authorized limits of release, which are usually lower than general dose limits and apply to exposure or discharge from all operated units on the site.

The above requirements for the new nuclear installation project for operating states of a nuclear installation are in conformity with documents IAEA [L. 269], ICRP [L. 254] and WENRA [L. 27] and [L. 270], which require that effluent limits specified by supervision are not exceeded.

Design basis accidents

Design basis accidents are such accidents for which it has been demonstrated, using conservative methodologies in the design, that radiological consequences are kept within the acceptance criteria.

Basic design basis accidents include for example events caused by the failure or disruption of civil structures, technological systems and equipment, operators' errors or events caused by external events that result in disruption of operational limits and conditions of safe operation and which may cause even limited damage to the fuel system.

In terms of radiation protection, the following radiological criteria were specified for these nuclear installation states (with occurrence frequency from 10^{-6} to 10^{-2} per year) in conformity with EUR [L. 264] document.

Contingent release of radioactive substances into the atmosphere must be limited so that:

- it does not cause the necessity to adopt any measures in a distance exceeding 800 m from the reactor,
- it has a very limited economic impact outside the nuclear installation

Design extension conditions

Design extension conditions are such accidents that are not considered for design basis accidents, but that are considered in the design process of the plant in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. According to WENRA [L. 270] document, they may include severe accidents.

Severe accidents that may result in early or substantial releases and are not included in this category must be practically excluded.

In terms of radiation protection, the following radiological criteria were specified for these nuclear installation states (with occurrence frequency lower than 10^{-6} per year) in conformity with EUR[L. 264] document.

Contingent release of radioactive substances into the atmosphere must be limited so that:

- it is not necessary to launch evacuation (exigent protection measure) in a distance greater than 800 m from the reactor. This measure is based on the value of the design (expected) dose in 7 days and at the time when significant radioactive releases may occur
- it is not necessary to launch temporary relocation as a delayed protection measure beyond the limit of 3 km from the reactor at any time. This measure is based on the value of the designed (expected) dose within 30 days caused by exposure from the contaminated earth's surface and inhaling due to re-suspension of aerosols
- it shall not necessary to launch permanent relocation as a delayed protection measure beyond the limit of 800 m from the reactor at any time. This measure is based on the value of the designed (expected) dose per 50 years caused by exposure from the contaminated earth's surface and inhaling due to re-suspension of aerosols. The doses caused by ingestion are not considered in the case of this measure
- Limited economic impact - restriction of food and crops consumption shall be limited by time and area

The above mentioned requirements for new nuclear installation design for accident conditions shall also be in conformity with WENRA [L. 27] and [L. 270] documents, which require that if an accident does not cause severe damage to the fuel system, it should not cause any radiological impact on the power plant surroundings or only minimal impact [101] (particularly such impact when it is not necessary to implement iodine prophylaxis, sheltering or evacuation - regulatory measures for distribution of food and animal feed may not be excluded). Following a severe accident when the fuel system is severally damaged, which may not be practically excluded, such design measures need to be adopted that the introduced population and environment protection measures are only time and area limited (and would not include for example permanent relocation of population, necessity of emergency evacuation from the vicinity of the power plant, limited sheltering of persons, any long term limitation of food consumption) and there is enough time for taking the given measures.

For accidents with severe damage to the fuel system when it is not possible to exclude early and substantial releases of radioactive substances, the design shall also include all technical and organizational means which the plant operator needs to

[101] [WENRA, 2009] (chapter 7, p. 13) - "no or only minimal radiological impact" – the intervention values are not reached which would result in launching of iodine prophylaxis, sheltering or evacuation. These intervention levels shall be used up to the fifth level of in-depth protection and be consistent with ICRP recommendation.

meet all its obligations given by law (Article 19 of Act No. 18/1997 Coll. [L. 2]). Introduction of respective protection measures shall be based on the criteria stipulated by the legislation of the CR, EU and IAEA and ICRP recommendations.

4.1.2 BASIC PRINCIPLES OF RADIATION PROTECTION

General principles of securing of radiation protection according to legal regulations of the EU, CR and according to ICRP and IAEA recommendations are based on basic principles of radiation protection.

Principles of radiation protection

In 1990 recommendations, ICRP stated principles of protection for radiation activities separately from intervention situations. These principles are further considered as the basis for the protection system. New ICRP recommendation [L. 254] forms a set of principles which may be used individually for planned situations, accident situations and for the existing exposure situations. The recommendation clarifies how to apply basic principles with sources and representative individuals, whereas the principles apply to the sources in all controllable situations. The first two principles are related to the source and are used in all exposure situations; the third principle is related to the individuals and is used in planned exposure situations.

- The principle of justification - any decision which changes radiation exposure situation should cause more benefit than harm. (In order to correct the caused harm, sufficient social benefit should be achieved by operation of the nuclear power plant.)
- Protection optimisation principle – possibility of development of exposure situation, number of exposed people and the size of individual doses - all these need to be kept as low as is reasonably possible considering economic and social aspects. (In order to prevent significant disparities during the optimisation process, doses or risks for individuals should be limited from a particular source by dose or risk optimisation limits.)
- The principle of application of dose limits - total dose to any individual from controlled sources in planned exposure situations (with the exception of medical exposure) should not exceed respective limits.
- In conformity with IAEA recommendations [L. 269], the application of the principle of protection optimisation and dose limits is reflected in safety measures, which need to be focused on such limitation of doses so that besides not exceeding authorized limits it is secured that the sum of annual radioactive releases into the environment shall be limited so that the actual dose in any year for any individual (including persons in remote sites and future generations) with high probability does not exceed any relevant dose limit even considering further possible designed exposures.

In the case of completion of NPP Temelín, the reasoning of the intention is based on securing of power security of the state. The use of the Temelín site is given by the approved Local Development Policy, and assessment of negative impacts against benefits is a part of assessment of the impact on the environment.

Optimisation of radiation protection shall be at design level solved by the project contractor and the achieved level assessed within the tender proceedings. At the level of management of the nuclear power plant the optimisation principle shall be in conformity with the legislation requirements of Decree No. 307/2002 Coll. [L. 4] applied in operating instructions for radiation activities, which shall be based on respective permissions of the regulatory body and within their approved documentation.

Using a monitoring programme together with operating instructions and care for a high culture of operation observance of respective limits shall be secured and controlled. Authorized limits shall secure even protection of the environment in terms of possible cumulative effects of introduction of radionuclides into the biosphere.

4.1.3 CONCEPT OF RADIATION PROTECTION FOR PLANNED AND ACCIDENT EXPOSURES

Planned exposure situations include:

- normal operation of the nuclear power plant (normal exposure) and
- exposures that may occur in the course of operation time / lifetime of the power plant in the case of abnormal operating conditions considered with probability greater than ca 10^{-2} (potential exposure)

Accident exposure situations are not planned situations; however, their occurrence cannot be excluded. Yet, procedures and measures shall be drawn up (in respective accident plans) to avert/mitigate the impacts of such accidents on people and the environment.

The principle of optimisation of protection based on assessment of contributions and costs and respecting of dose constraints shall be applied in the planned and accident exposure situations.

Population and environment exposure during planned exposure situations

The radiation protection concept in relation to population and the environment requires the following:

- In conformity with CR legislation, population exposure in planned exposure situations is subject to annual limit of the effective dose from external exposure and the load of effective dose corresponding to the annual value equal to 1 mSv. However, the causes of exposure considered are limited only to releases of radioactive substances to which maximum values of optimization limits apply. (average effective dose of 250 µSv per calendar year with respective critical population group of which 200 µSv for releases into the atmosphere and 50 µSv for releases into watercourses)
- According to recommendations from ICRP [L. 254], it is possible to consider that under special circumstances even a higher value of the effective dose is admissible in one year provided that the average across the defined five-year period does not exceed 1 mSv per year. (These higher - the so-called potential - exposures may occur only in abnormal operating conditions or as a result of accidents)
- However, in the case of planned exposure situations in normal operation of the NPP, the assumption is that exposure of persons does not exceed the specified authorized limits, which cause only insignificant increasing above the values of doses of natural background and are significantly lower than the maximum value for the dose constraint and secure thus a strict protection

level and correspond to situations in which individuals receive doses which are not a direct contribution to them but a contribution to the society

- Substances, materials and objects containing radionuclides or surface polluted by radionuclides shall be handled in such a way so that in any calendar year, the collective effective dose associated with introduction into the environment does not exceed 1 Sv and with any individual the sum effective does not exceed 10 µSv
- The nuclear installation shall be designed in such a way so that the production of radioactive waste and releases causing radiation strain on population is as low as reasonably achievable

Occupational exposure

Occupational exposure occurs in planned exposure situations when radiation protection is planned before exposure occurs and where the scope and degree of exposure may be reasonably forecast. The optimisation range for planned exposure situations from 1 to 20 mSv corresponds in conformity with ICRP recommendation [L. 254] to the situation of workers who are informed of the associated health risk and take it voluntarily in the given wage conditions.

Control of exposures for planned exposures in this range shall be secured by design of workplaces and further conditioned by introduction of the system of personal monitoring and corresponding training of workers.

Basic arrangement of areas inside the new nuclear installation according to EUR [L. 264] requirements is described by Fig. 62

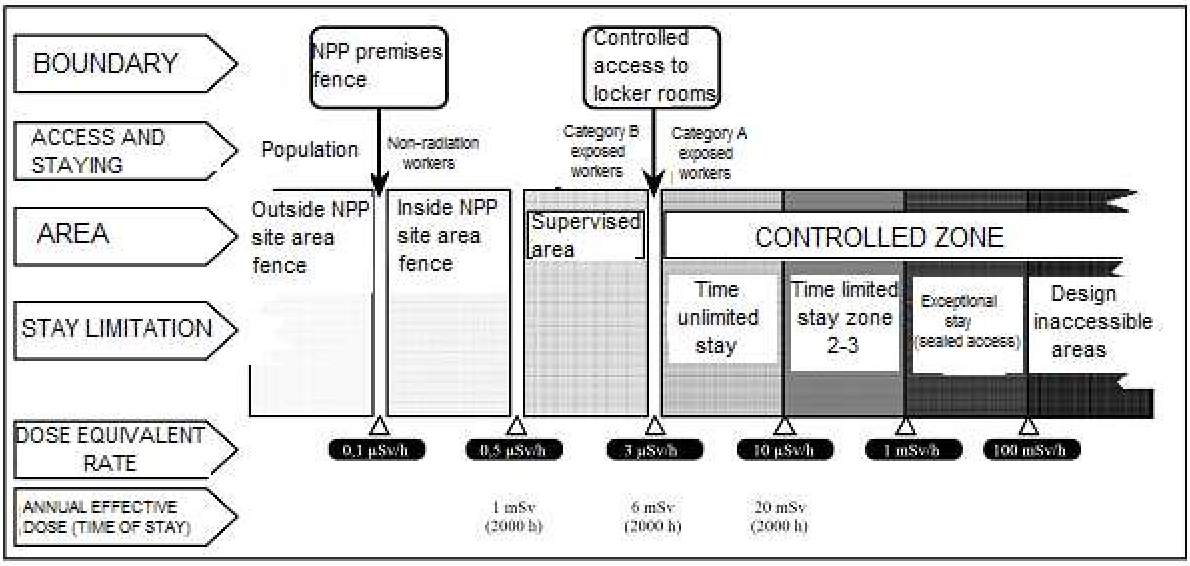

Fig. 62 Arrangement of nuclear installation premises according to EUR requirements [L. 264]

The diagram above corresponds to legislation requirements for delimitation of the monitored and controlled zone and the instructions for more detailed articulation of the controlled zone which is primarily an organisational measure and shall result in desirable limitation of personnel exposure.

For occupational exposure in planned exposure situations and in line with legislation and ICRP recommendation, the considered limit of annual effective dose from external exposure and the committed effective dose corresponding to the annual amount is 20 mSv, or an average limit across the defined five-year period (100 mSv in 5 consecutive years), provided the exposure not exceed 50 mSv in any single year.

By applying optimisation procedures both in normal operation and specific cases (e.g. during activities associated with maintenance and repair of equipment or activities implemented in otherwise not attended areas) it is secured that exposure of staff is limited to the level under the specified limits (IAEA recommendations state 5 mSv/r as an objective) as is possible in the given situation / during the given activity.

Irradiation due to radiation accidents

In the case of exposure accident, the assumption is that:

- Doses higher than 100 mSv mean increased possibility of creation of deterministic effects and significant risk of stochastic effects (cancer). For these reasons and according to ICRP [L. 254] recommendations the dose of 100 mSv received at one time or in the course of the year is regarded as the maximum value for the reference level. A protective measure is almost always reasoned by a dose nearing 100 mSv and exposures over 100 mSv received either at one time or in the course of the year may be substantiated only in extreme situations (e.g. saving of life or prevention of serious disaster)
- Doses between 20 mSv and 100 mSv correspond to unusual and limit situations, particularly in conditions of a severe accident when measures are not available for radical reduction of doses. Doses of representative persons from members of the public in the range from 1 mSv to 20 mSv correspond particularly to situations after very unlikely events, for which though, adequate protection measures are available

When assessing the design, adequately conservative designs shall be used for investigation of probability and consequences of a sufficiently representative spectrum of initiating events. Safety analyses that shall be included in all further stages of safety documentation shall verify adequacy of the technical solution with regard to the degree of risk associated with operation of the plant in normal, abnormal, and accident conditions.

The decision of acceptability of potential exposures must take into account the probability of exposure occurrence and its scope. In some circumstances, the decisions may be based on divided assessment of probability of occurrence of exposure and its scope.

In the case of preliminary assessment of design safety based on the achievement of the above-mentioned objectives EUR [L. 264] uses a very simplified model allowing assessment of the level of type design in terms of radiation risks for the surroundings without knowing particular conditions of the site. This model considers only two basic dispersion situations - the situation with better dispersion conditions (high elevation release) and the situation with worse dispersion conditions (the so-called ground release). It further operates with a very simplified inventory of the source member, which presents maximally 9 representatives of the most significant radionuclides. For each of these radionuclides, two coefficients are given (one for the high-elevation and one for the ground release)) replacing conversion factors for all exposure pathways. Criterion values are set so that they may be assigned to the intervention levels for urgent and longer term protection measures.

In the case of guidance assessment of contingent scope of a severe contamination of the environment which might also mean limiting of agricultural production in the impacted territory, EUR [L. 264] uses even simpler criteria consisting in limiting of activities of two or three radionuclides that are most dominant in this regard.

As the supporting documents of the particular design shall be available for safety analyses, which shall be a part of the ISAR, it shall be possible to assess the radiation consequences in much greater detail even with regard to meteorological, geographical and demographic conditions of the Temelín site.

However, even with the level of the current knowledge it is possible to state that the concept of considered intention is in terms of radiation protection selected so that the radiation consequences even of those least probable radiation accidents with the greatest considerable impacts on the population have been limited mainly to the protection zone territory, i.e. the land on which no permanent habitation structures are present.

4.2 INTRODUCTION OF RADIONUCLIDES INTO THE ENVIRONMENT

4.2.1 REASONING FOR INTRODUCTION OF RADIONUCLIDES INTO THE ENVIRONMENT

Detailed substantiation of introduction of radionuclides into the environment shall be a part of the documentation for the licence according to Article 9 (1)(h) of Act No. 18/1997 Coll. [L. 2]. This substantiation shall be based on design data for the particular installation and shall be complemented with:

- specification of radionuclide composition and activities of radionuclides introduced into the environment
- evaluation of exposure of critical group of the population by radioactive releases (based on effective doses for representative persons)
- analysis of possible accumulation of radionuclides in the environment in the case of their long-term releasing

The issues of radioactive releases was already subject to the assessment of impact on the environment according to Act No. 100/2001 Coll. [L. 255], which is a requisite condition for issuing of the permission for placement of nuclear installation and shall also be taken into account in respective quality assurance programmes as stipulated by Article13 (4),(5) of Act No. 18/1997 Coll. [L. 2].

4.2.2 GENERAL PRINCIPLES FOR INTRODUCTION OF RADIONUCLIDES INTO THE ENVIRONMENT

Releases into the atmosphere and into watercourses shall be monitored so that continuous balance measuring of all radionuclides which significantly contribute to exposure of the public is secured together with continuous measuring of representative radionuclides which may quickly indicate deviations from normal operation. The measured values shall be used for continuous monitoring of use of the authorized limit and in the case that deviation values are ascertained from normal operation, they shall be the stimulus for adopting relevant measures.

4.2.3 DISCHARGE INTO THE ATMOSPHERE

4.2.3.1 RADIONUCLIDE COMPOSITION AND ACTIVITIES OF GAS DISCHARGES

As it is practically impossible in nuclear power plant operation to prevent ingress of certain fractions of formed radionuclides into the systems of technological exhausting and the rooms inside the checked zone and the filtration units of respective systems of exhaust air conditioning may not achieve absolute efficiency, it is necessary to accept presence of radionuclides in gas discharges in the degree corresponding to the current state of technology, economic aspects and limiting legislation requirements.

The spectrum and quantity of radionuclides which may be practically expected in gaseous discharges follow from the principle of power engineering use of uranium enriched by ^{235}U isotope, PWR type reactor concept and the considered annual production of electric energy.

The sources of gaseous radioactive discharges into the air are technological systems of the primary part, systems of processing and treatment of radioactive waste and exhaust air conditioning of the controlled zone buildings.

Products of ^{235}U fission penetrate into coolant by diffusion in micro-cracks in the fuel coverage (or possibly through leaks in a damaged fuel cell) and formation of fission products on the external cell surface may not be completely excluded either if it has been contaminated in production even by a trace amount of fuel. Releasing of products from the external surface of coverage may prevail in the initial period of operation. Later in operation micro-defects of coverage start to occur more frequently and subsequently, due to ageing of material, a partial breach of hermetic sealing of fuel may not be excluded. The most significant gaseous fission products are krypton and xenon isotopes. Radioisotopes of iodine, caesium and other volatile elements are radiologically significant but easily separable fission products.

Besides fission products, activation products are formed in the area of the reactor active zone and in its immediate vicinity due to nuclear reactions with neutrons. The most significant one in terms of population and environment irradiation in normal operation is ^{14}C radionuclide which is formed primarily by reactions of $^{17}O(n,\alpha)$ ^{14}C and ^{14}N (n,p) ^{14}C. Another significant radionuclide is ^{41}Ar, which is formed by reaction of ^{40}Ar (n,γ) ^{41}Ar.

Based on available preliminary documents, it is assumed that the annual discharge of radionuclides most significant for radiation from two nuclear units (even in the case of the highest considered power) shall not exceed maximum values stated as follows Tab. 157.

Comment: With regard to the relation of yield of fission products to the overall power, the gross power is stated, which due to service consumption is by ca 100 MWe higher than net power.

Tab. 157 Preliminary estimate of maximum annual discharges of radionuclides into the atmosphere from operation of nuclear units with gross power of up to ca 2 x 1700 MWe

radionuclid e	annual discharge [Bq]	radionuclide	annual discharge [Bq]
H-3	2.6E+13	Ru-103	5.9E+06
C-14	7.0E+11	Ru-106	5.8E+06
Ar-41	2.5E+12	Sb-125	4.5E+06
Cr-51	4.5E+07	I-131	8.9E+09
Mn-54	3.2E+07	I-133	3.0E+10
Fe-59	5.6E+06	Xe-131m	2.6E+14
Co-58	1.7E+09	Xe-133m	1.3E+13
Co-60	6.4E+08	Xe-133	6.4E+14
Kr-85m	1.1E+13	Xe-135m	1.0E+12
Kr-85	2.5E+15	Xe-135	8.9E+13
Kr-87	3.9E+12	Xe-138	8.9E+11
Kr-88	1.3E+13	Cs-134	1.7E+08
Sr-89	2.2E+08	Cs-136	6.3E+06
Sr-90	8.9E+07	Cs-137	2.7E+08
Zr-95	7.4E+07	Ba-140	3.1E+07
Nb-95	1.9E+08	Ce-141	3.1E+06

Conservatism of the above data is given by considering the maximally admissible design parameters of radioactive media in technological systems of the primary circuit and the theoretical atmosphere discharges derived from them. In reality, it may be expected that the long-term average of operation discharges shall be close to the discharges of the existing PWR units of a similar concept.

In order to verify reality of the preliminary data of discharges from the new units, it is possible to compare the values stated in Tab. 157 with the following estimate performed using normalized discharges taken over from UNSCEAR 2008 [L. 267] statistical data.

Tab. 158 Estimate of annual discharges based on normalized discharges of existing PWR units operated around the world

	Normalized discharge Bq/GWr (according to UNSCEAR 2008)	Annual discharge in Bq with gross power of 3400MW and operation of ca 8000 hours
Noble Gases	1.10E+13	3.42E+13
Tritium	2.10E+12	6.53E+12
I-131	3.00E+08	9.33E+08
C-14	2.20E+11	6.84E+11
Aerosols	3.00E+07	9.33E+07

The trend of development of radioactive discharges from units with PWR is documented by Fig. 63. It is possible to declare that there is a gradual marked drop of discharge of noble gases, however, virtually constant with radiologically significant C-14. Therefore with new units of a similar concept it is possible to expect decreasing of overall activity of gaseous discharges but without significant impact on exposure of representative persons.

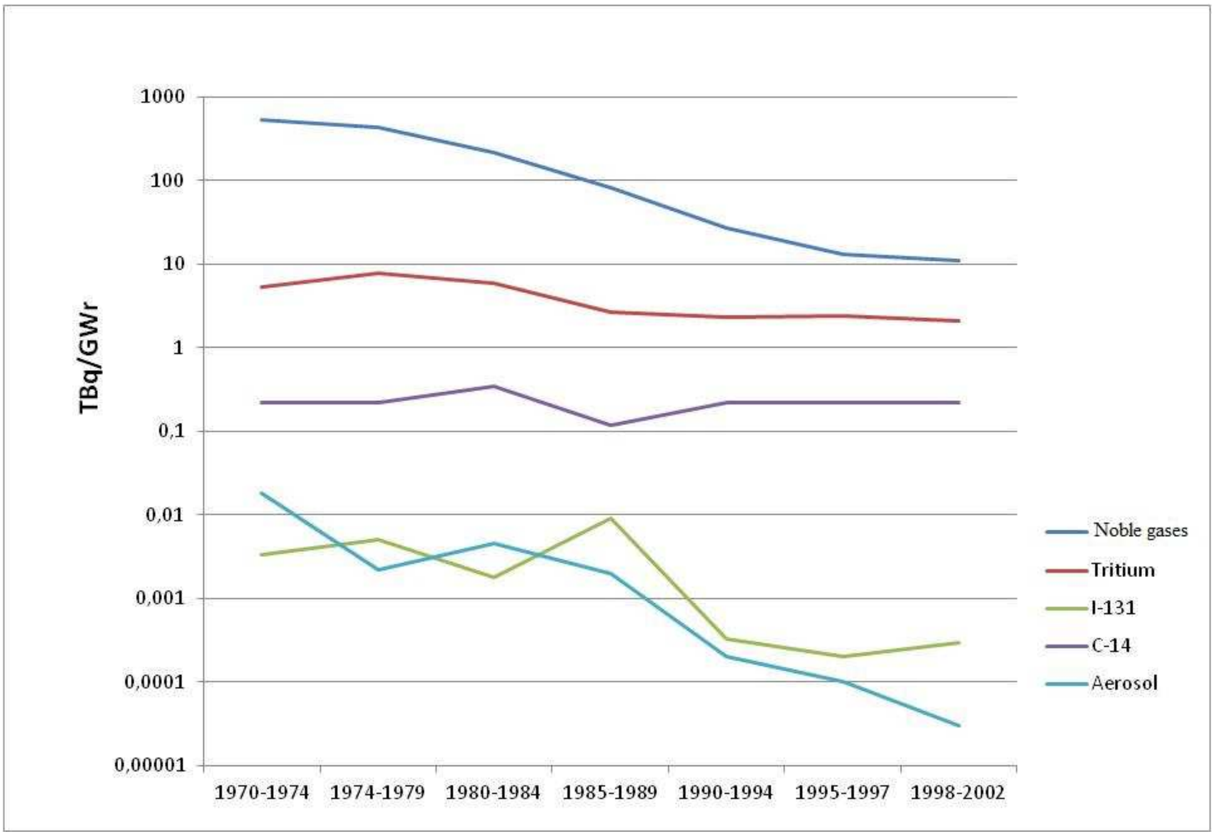

Fig. 63 Radionuclide discharges from PWR units operated around the world between 1970 and 2002 into the atmosphere in TBq per GWr of generated power (source: Sources and Effects of Ionising Radiation, UNSCEAR 2008)

4.2.3.2 PRELIMINARY ASSESSMENT OF RADIATION CONSEQUENCES FOR POPULATION AND THE ENVIRONMENT

The preliminary assessment of consequences of introduction of radionuclides into the atmosphere is based on conservative inventory of annual discharges as stated in Tab. 157.

The estimate of the sum of effective doses of external exposure and the effective committed doses from internal contamination of a representative person corresponding to the reception from annual discharges of radionuclides into the atmosphere was performed as the sum of products of activities of radiologically significant radionuclides and transfer coefficients *h* specified for the conditions on the site by the existing decision of SÚJB ref. No. 28718/2007 for the NPP Temelín.

Tab. 159 Estimate of annual effective dose of a representative person due to discharges of radionuclides into the atmosphere

radionuclide	annual discharge [Bq]	h [Sv/Bq]	annual effective dose [Sv]
H-3	2.6E+13	5.20E-22	1.35E-08
C-14	7E+11	1.93E-19	1.35E-07
Ar-41	2.5E+12	1.43E-21	3.58E-09
Cr-51	4.5E+07	8.70E-20	3.92E-12
Mn-54	3.2E+07	2.05E-17	6.56E-10
Fe-59	5.6E+06	4.17E-18	2.34E-11
Co-58	1.7E+09	5.60E-18	9.52E-09
Co-60	6.4E+08	3.54E-16	2.27E-07
Kr-85m	1.1E+13	2.73E-21	3.00E-08
Kr-85	2.5E+15	4.55E-23	1.14E-07
Kr-87	3.9E+12	1.15E-20	4.49E-08
Kr-88	1.3E+13	3.24E-20	4.21E-07
Sr-89	2.2E+08	1.11E-19	2.44E-11
Sr-90	8.9E+07	5.80E-17	5.16E-09
Zr-95	7.4E+07	3.99E-18	2.95E-10
Nb-95	1.9E+08	2.22E-18	4.22E-10
Ru-103	5.9E+06	1.77E-18	1.04E-11
Sb-125	4.5E+06	3.72E-17	1.67E-10
I-131	8.9E+09	1.19E-18	1.06E-08
I-133	3.0E+10	2.04E-19	6.12E-09
Xe-133	6.4E+14	5.95E-22	3.81E-07
Xe-135m	1.0E+12	3.68E-21	3.68E-09
Xe-135	8.9E+13	4.44E-21	3.95E-07
Xe-138	8.9E+11	8.70E-21	7.74E-09
Cs-134	1.7E+08	8.65E-17	1.47E-08
Cs-137	2.7E+08	1.47E-16	3.97E-08
Ba-140	3.1E+07	2.24E-19	6.94E-12
Ce-141	3.1E+06	3.82E-19	1.18E-12
Total			1.86E-06

With regard to the fact that for assessment of radiation consequences for the population and the environment it is also necessary to have an analysis of possible accumulation of radionuclides in the environment in the case of their long-term discharge, preliminary analyses were performed taking into account (besides probable meteorological conditions) the effect of gradual increasing of the deposit in the course of planned operation of the installation.

The calculations were performed using the NORMAL version 02 program whereas for the assessed period real meteorological data from the period 2000 – 2006 were used together with expert opinion specified parameters of the vent stack (altitude at the foot 507 metres above sea level, height 100 m, orifice diameter 1.6 m, vertical discharge speed 12.5 m/s).

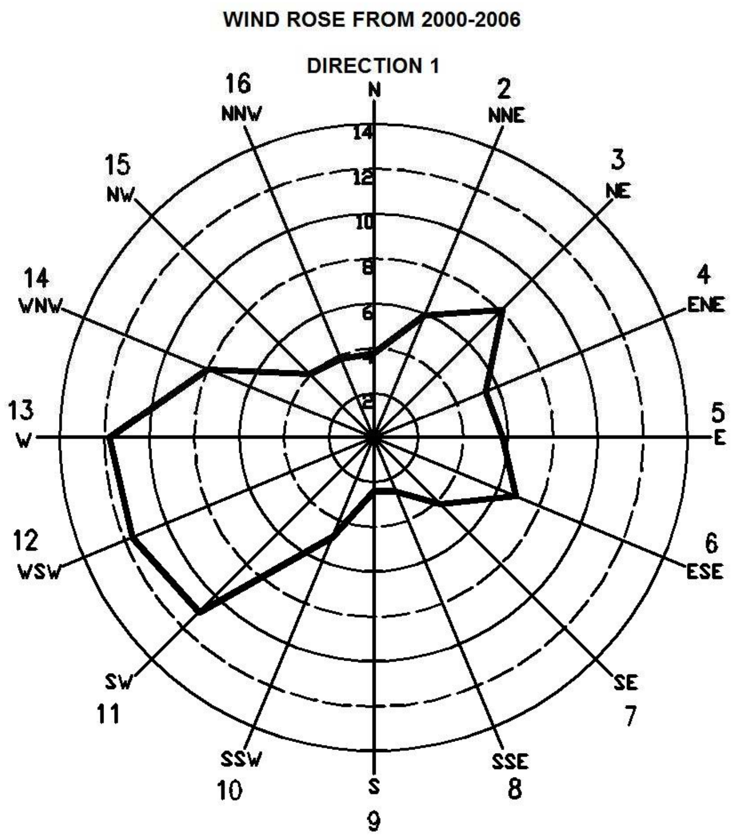

Fig. 64 Wind rose used for assessment of impact of discharge into the atmosphere

More conservative assumptions were used when calculating ground volume activities, deposition on the ground, deposition input (due to long term weighed factors of dry and wet fallout), annual doses from external exposure and committed effective doses from annual income of radionuclides, particularly:

- The Hosker model was used when describing dispersion - it gives a narrower and therefore more active plume for the respective area of contact with earth
- Thermal lift of the plume is not considered (the plume is closer to the earth surface)
- Conservative values were selected in conversion factors for ingestion depending on type of absorption in the digestive organs according to Decree No. 307/2002 Coll. [L. 4]
- When calculating dry fallout and deposition, conservative (higher) values were used for the atomic form of iodine and aerosol form of radionuclides
- When calculating annual depositions, radioactive decomposition and contingent subsidiary products were considered. Deposition on the earth's surface in the given year is calculated as a sum of deposition from constant deposition input in the given year plus deposition in previous years minus radioactive decomposition and the processes of removal of activity from the earth's surface. The time course of deposition of an activity therefore includes disintegration and migration (expressed by an effective diminishing constant selected based on expert recommendations)
- The leaf way in the particular year plus transport through a root are considered for annual intakes into the human organism by ingestion. The root

way is based on the activity gradually deposited during the particular year (with constant deposition input) plus volume activity in the root zone due to settling in previous years. When calculating annual intake of an activity in the particular year, fixing of radionuclides in the root zone was also included when radionuclides fixed in the ground are not available for further transport of the activity into plants via the root way

- Long-term deposition nears the balanced state more slowly than the annual intake of activity by an individual in the following years. The reason is considering of earth fixation of radionuclides with a long half-life, and further, the prevailing effect of transport of radionuclides by the leaf way compared to the root way
- The consumption basket was considered in a conservative way: local production - local consumption
- Resulting values therefore present maximum hypothetically possible levels of planned exposures when the person stays on the given site all the time in the open (without correction due to screening by buildings, without location factors and persistence coefficients) and consumes only locally grown products

The following table Tab. 160 states the results of calculation of the annual effective dose of an individual from the age category of adults located in direction 3 (northeast), where according to the wind rose, representative persons may be considered even in the following years. For this direction the maximum theoretically achievable exposure situation may be expected at the end of the plant's service life (ca after 60 years), when accumulation effects of settling of radionuclides in the environment might show.

According to the preliminary assessment, the most exposure representative person shall not receive more than 5 micro Sv/r – see the following chart using results of calculations performed within supporting materials for Documentation of Environmental Impact Assessment [L. 29].

Tab. 160 Theoretical maximum annual individual effective doses from external exposure and effective committed doses from annual intake after 60 years of operation of 2 units with the gross unit output up to ca 1700 MWe on the border of the protection zone in the direction of the greatest impact

Distance	Cloud dose	dose from actual annual deposit	dose from accumulated deposit 60th year	committed inhalation dose	committed ingestion dose	committed inhalation re-suspension dose	Total approximately
approx. kM	µSv	µSv/year	µSv	µSv	µSv	µSv	µSv
3	0.6	0.07	4	0.08	0.1	10^{-6}	< 5

The above mentioned facts make it clear that the radiation impacts of discharges into the atmosphere on the surrounding population shall be insignificant under normal operation. (The contribution to exposure from natural sources shall be maximally at the level of per milles.) Therefore it is possible to consider the legislation requirements for securing of the radiation protection as fulfilled.

Similarly, insignificant impacts on the environment are in conformity with the approach of ICRP [L. 254] and the protection of the environment may be regarded as sufficiently secured.

The values of the effective dose of representative persons of various age categories due to practical operation shall be calculated by model calculation by SÚJB approved programme tool (currently RDETE) and using real meteorological data ČHMÚ – Praha for the assessed period, i.e. the past operating year.

4.2.4 DISCHARGE INTO WATER

4.2.4.1 RADIONUCLIDE COMPOSITION AND ACTIVITIES OF LIQUID DISCHARGES

All waste water is created in a controlled zone, i.e. water with possible occurrence of radioactive substances, therefore they shall be controlled and regulated as the already operated NPPs in order to exclude inadmissible release of radioactive substances into the environment. Systems of separate sewerage and water treatment plants of this waste water shall be established to intercept radioactive substances. The treated waste water shall be collected in control tanks and released into streams subject to a check proving conformity with conditions of the decision of respective state administration and surveillance authorities. In the case of a negative result of the implemented check, the content of the control tank shall be further processed to comply with these conditions.

In terms of environmental impact, tritium is the dominant radionuclide for normal operation and its production is among other things heavily dependant on the chemical mode of the primary circuit and the type of fuel. Unlike other radionuclides, its separation would be technologically very demanding and considering the associated costs contrary to the optimisation of radiation protection, and therefore this radionuclide composes more than 95% of discharged radioactivity into watercourses.

When operating both units of the new nuclear source with outputs within the considered range, the preliminary estimate is that the annual discharges of radionuclides shall not exceed the maximum values stated in Tab. 161. The values stated in the table are based on the available design data. Conservatism of the above data is given by considering the maximally admissible design parameters of radioactive media in technological systems of the primary circuit (limiting activities) and the theoretical watercourse discharges derived from them. In reality, it may be expected that the long-term average of operation discharges shall be close to the discharges of the existing PWR units of a similar concept.

The trend of development of liquid radioactive discharges from units with PWR reactors is documented by Fig. 65.

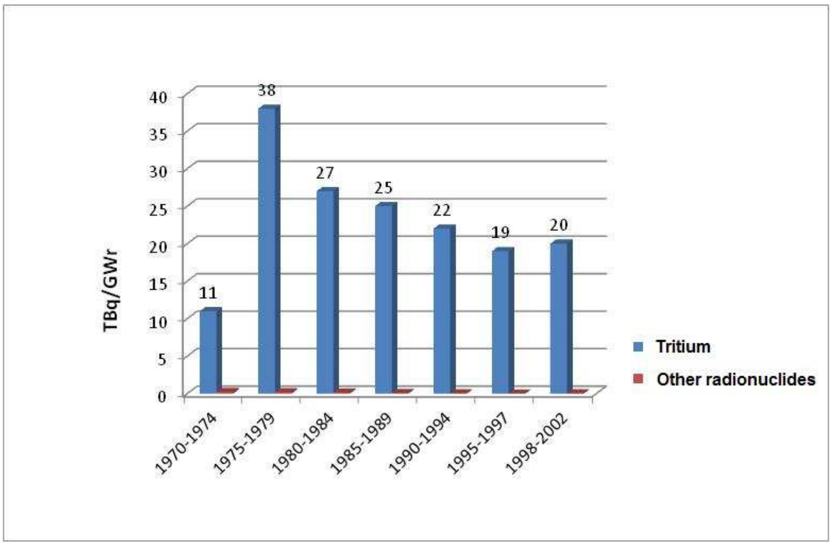

Fig. 65 Liquid radionuclide discharges from PWR units operated around the world between 1970 and 2002 in TBq per GWr of generated power (source: Sources and Effects of Ionising Radiation, UNSCEAR 2008)

Tab. 161 Preliminary estimate of maximum annual discharges of radionuclides into watercourses from operation of nuclear units with gross output of up to ca 2 x 1700 MW

radionuclide	annual discharge [Bq]	radionuclide	annual discharge [Bq]
Na-24	4.5E+08	Ag-110m	1.3E+08
P-32	1.3E+07	Ag-110	1.0E+07
Cr-51	4.4E+08	Sb-124	5.88E+07
Mn-54	7.0E+08	Te-129m	8.9E+06
Fe-55	6.42E+08	Te-129	2.3E+07
Fe-59	1.7E+08	Te-131m	2.3E+07
Co-58	7.3E+08	Te-131	5.6E+06
Co-60	1.0E+09	I-131	2.5E+09
Ni-63	6.91E+08	Te-132	3.6E+07
Zn-65	3.0E+07	I-132	8.9E+07
W-187	3.4E+07	I-133	2.6E+09
Np-239	4.3E+07	I-134	1.1E+09
Rb-88	2.1E+09	Cs-134	8.9E+08
Sr-89	1.1E+07	I-135	3.7E+08
Sr-90	2.3E+06	Cs-136	1.6E+09
Sr-91	5.9E+06	Cs-137	1.3E+09
Y-91m	3.7E+06	Ba-137m	9.3E+08
Y-91	6.7E+06	Ba-140	4.3E+08
Y-93	2.7E+07	La-140	5.9E+08
Zr-95	9.6E+07	Ce-141	2.1E+07
Nb-95	1.5E+08	Ce-143	4.5E+07

radionuclide	annual discharge [Bq]	radionuclide	annual discharge [Bq]
Mo-99	1.3E+08	Pr-143	9.6E+06
Tc-99m	1.3E+08	Ce-144	4.1E+08
Ru-103	3.6E+08	Pr-144	2.3E+08
Rh-103m	3.6E+08	other	1.5E+06
Ru-106	5.4E+09	Total	3.15E+10
Rh-106	5.4E+09	H-3	1.2E+14

In order to verify the feasibility of the preliminary data of the discharges from the new units it is possible to compare the values stated in Tab. 161 with the following estimate performed using normalized discharges taken over from UNSCEAR 2008 [L. 267] statistic data.

Tab. 162 Estimate of annual liquid discharges based on normalized discharges of existing PWR units operated around the world

	Normalized discharge Bq/GWr (according to UNSCEAR 2008)*	Annual discharge in Bq with gross output of 3400 MW and operation time of ca 8000 hours
Tritium	2.00E+13	6.22E+13
Other radionuclides**	1.10E+10	3.42E+10

** Last published data from 1998 - 2002*

*** The quoted source does not state more detailed specification of other radionuclides. This is, however, a sum based on the data published by operators of individual NPPs complemented with expert estimates in the case of unavailable data.*

The comparison makes clear particularly the conservatism of tritium values, which are further used for assessment of radiation consequences of operation. Further, the diagram (see **Fig. 65**) makes it clear that historically, the greatest differences were in annual production of tritium water.

4.2.4.2 PRELIMINARY ASSESSMENT OF RADIATION CONSEQUENCES FOR POPULATION AND THE ENVIRONMENT

With regard to the position of new units in the premises of the existing NPP Temelín, the future common outlet of liquid discharge from all 4 units to the Vltava in the Kořensko profile is assumed.

Within detailed specification of technological solution of new units and the resulting requirements for introduction of radionuclides into the environment, the regime of discharge of water from NPP Temelín premises shall be designed so as to minimize negative impact of the Vltava River. The final technological solution shall result from the conditions of decisions of respective state administration and surveillance authorities and shall accept requirements not only regarding the quality of waste water but also regarding the overall volume, flow-through volumes, and the method of monitoring .

The estimate of the sum of effective doses of external exposure and the load of effective doses from internal contamination of a representative person corresponding to the reception from annual discharges of radionuclides was performed as the sum of products of activities of radiologically significant radionuclides stated in Tab. 163 and transfer coefficients *h* specified for the conditions on site by the existing decision of SÚJB ref. No. 26161/2009for the NPP Temelín.

The implemented preliminary estimate of impact of introduction of radionuclides into watercourses shall lead to the conclusion that the theoretical annual individual effective dose of a representative person should not exceed the level of ca 6 µSv.

With regard to the fact that the dose optimisation limit for the members of the public shall be complied with a substantial margin, the protection of water life according to the recommendations of ICRP [L. 254] may also be regarded as sufficiently secured.

Tab. 163 Estimate of annual effective dose of a representative person due to discharges of radionuclide into watercourses

radionuclide	annual discharge [Bq]	h [Sv/Bq]	annual effective dose [Sv]
H-3	1.20E+14	4.17E-20	5.00E-06
Na-24	4.50E+08	5.34E-19	2.40E-10
Cr-51	4.40E+08	2.17E-18	9.55E-10
Mn-54	7.0E+08	3.63E-17	2.54E-08
Fe-59	1.70E+08	5.79E-17	9.84E-09
Co-58	7.30E+08	7.92E-17	5.78E-08
Co-60	1.00E+09	6.98E-16	6.98E-07
Zn-65	3.00E+07	3.77E-17	1.13E-09
Sr-89	1.10E+07	9.58E-18	1.05E-10
Sr-90	2.3E+07	1.68E-16	3.86E-9
Zr-95	9.60E+07	1.12E-17	1.08E-09
Nb-95	1.50E+08	4.22E-19	6.33E-11
Mo-99	1.30E+08	3.57E-19	4.64E-11
Ru-103	3.60E+08	3.01E-18	1.08E-09
Ru-106	5.40E+09	1.04E-17	5.62E-08
Ag-110m	1.30E+08	2.31E-17	3.00E-09
Sb-124	5.88E+07	1.41E-18	8.29E-11
I-131	2.50E+09	8.02E-17	2.01E-07
I-133	2.60E+09	9.58E-18	2.49E-08
Cs-134	8.90E+08	8.32E-17	7.40E-08
Cs-137	1.30E+09	1.06E-17	1.38E-08
Ba-140	4.3E+08	1.65E-19	7.10E-11
La-140	5.90E+08	1.85E-19	1.09E-10
Ce-141	2.10E+07	6.05E-18	1.27E-10
Ce-143	4.50E+07	8.96E-19	4.03E-11
Ce-144	4.1E+08	1.19E-17	4.88E-09
Total			**6.18E-06**

Values of the effective dose of representative persons of various age due to real operation shall be calculated by model calculation approved by a SÚJB programme tool (currently RDETE) and using the real average value of flow rate in the preceding period in the profile Vltava - Kořensko (the level of ca 50 m^3/s may be assumed).

4.2.5 OTHER SOLID MATERIALS AND OBJECTS CONTAINING RADIONUCLIDES OR MATERIALS CONTAMINATED BY THEM

General characteristics

The source of contamination of various objects (clothes, protection equipment, defective further unusable parts of installation, etc.) is the contact with active media – particularly water of the primary circuit. In addition to this randomly and possibly irregularly generated waste, the regular waste generation from the filters of active air-conditioning systems, activated measuring sensors and cartridges with control samples is expected. Further, following long-term operation, individual cases of handling of large objects and waste are assumed - particularly during reconstructions or more demanding repairs of equipment. The irradiated and spent fuel is in terms of radiation protection subject to the requirements for radioactive waste, however, before it is officially declared as radioactive waste, it is not radioactive waste by law.

A monitoring programme needs to be drawn up for workplaces with sources of ionising radiation and the workplaces shall be equipped with the corresponding means of radiation inspection in order to prevent unauthorized discharge of radionuclides outside the workplace and that sorting by type and contamination level is possible together with other possibilities of use of monitoring materials and objects.

The decisive criterion for separating of materials and objects which it is possible to handle in a normal way from the radioactive waste, to which special rules shall apply, is the non-exceeding of the discharge level.

Discharged solid materials and objects

In conformity with Article 9(1)(h) of Act No. 18/1997 Coll. [L. 2] and Article 57(1)(b) of Decree No. 307/2002 Coll. [L. 4] the process for monitoring of solid substances and objects taken out of the controlled zone or otherwise introduced into the environment shall be submitted to the State Office for Nuclear Safety as a part of approved documentation.

Besides monitoring methods, the design shall contain the proof that with regard to the considered radionuclide contaminant composition the proposed procedure and instrumentation secure a sufficient level of radiation protection of the population and the environment.

The waste released in this way then shall not be subject to any regulatory measures in terms of radiation protection and it shall be further treated in conformity with the waste disposal act. The total amount and composition of this waste is insignificant compared to the other operation waste of the power plant and from the point of view of Act No. 18/1997 Coll. [L. 2] and its implementation decrees, it shall not be monitored further.

Radioactive waste

The materials that do not conform to the stipulated criteria and whose decontamination shall be technically impossible or economically disadvantageous shall be further treated according to the rules for treating of radioactive waste.

The preliminary estimate of production of low and medium radioactive waste with volume after treatment for storage is up to ca 240 m^3 per year. Based on the experience of operated NPPs, ca 20-30 % of the overall amount of waste may be included in the category of waste with medium activity. The remaining part may be classified as low-activity waste. The stated value of waste to be stored shall depend on the used processing technologies and shall be further specified in further stages of documentation according to the documents from the contractor of the particular selected technology. The uncertainty of data given by the level of knowledge of the planned technology shall not affect the assessment of the impact of operation of the designed equipment on population and the environment because:

- throughout the storage of radioactive waste in the nuclear power plant premises, sufficient barriers shall be secured to prevent contamination of the surrounding areas
- during transport to the repository, corresponding cover sets shall be used and the transport shall be implemented in conformity with the issued permission
- the repository is designed so as to provide long-term insulation of stored radioactive waste even after termination of operation which is documented by safety documentation of this separate nuclear installation

Preliminary assessment of radiation consequences for population and the environment

The released materials and objects may be handled without causing radiation threat to the concerned persons and the environment. Provided the above requirements are met, it may be preliminarily stated that the below mentioned internationally acknowledged criteria shall be respected (considering the specific characteristics of the site where the release is implemented):

- no person shall receive an individual effective dose exceeding 10 μSv per year due to discharge into the environment
- discharge into the environment shall not cause a greater annual collective effective dose than 1 Sv

4.3 PRELIMINARY ASSESSMENT OF WORKING ENVIRONMENT AND IMPACT ON EMPLOYEES

The ETE3,4 design shall in basic safety approaches accept the requirements of EU and CR legislation and requirements based on EUR [L. 264] document and radiation protection philosophy based on ICRP recommendation [L. 254] and monitor achieving of objectives in terms of level of radiation protection of workers:

- average annual effective individual dose < 5 mSv
- annual collective effective dose < 0.5 Sv/unit

The above stated target values taken over from EUR [L. 264] into the Tender Documentation are fully in conformity with legislation requirements and create prerequisites assuring that the use of limits shall be at the lowest reasonably achievable level as in the case of the already operated units. The current practice shows that the average annual effective doses of workers range between 0.3 and 0.5 mSv and that the highest exposures (sporadically up to 20 mSv) occur particularly at specific activities secured by contractual external workers as already mentioned in Section 4.1.3. Therefore these shall be the activities to which the principle of optimisation of exposure of workers shall be applied – by organization-technical preparation of work – maintenance and repairs and expert preparation of workers securing these activities with increased radiation risk with the objective to maintain the work culture minimally at the existing level in NPP Temelín. The preliminary assessment of professional exposure leads to the following conclusions:

- It is feasible that annual individual effective doses in normal operation for the majority of radiation workers are below 5 mSv.
- Special work procedures shall be prepared in advance for workers performing exceptional, specific operations so that by using the ALARA principle, individual and collective doses are so low with regard to the limit as allowed by the implementation of the given activity/operation

5 CONCEPT FOR SAFE TERMINATION OF OPERATION

5.1 PRINCIPLES FOR SAFE TERMINATION OF OPERATION

According to Decree No. 185/2003 Coll. [L. 260], termination of operation shall mean a set of activities leading to termination of use of a nuclear installation or workplace, or to using for different purposes other than those approved under Section 9(1) of Act No. 18/1997 Coll. [L. 2]. Based on this decree, the termination of NPP operation shall be included into the decommissioning process as a separate stage. The problems related to decommissioning shall be dealt with and specified in the presented documents for the issue of a licence for the individual activities under Section 13(3) c) of the Atomic Act during the whole process of preparation, implementation, commissioning and operation of the NPP in accordance with the requirements laid down in Act No. 18/1997 Coll. [L. 2].

All requirements laid down in Act No. 18/1997 Coll. [L. 2] and the related legal decrees (Decree No. 185/2003 Coll. [L. 260] and other applicable acts and decrees, in particular Decree No. 307/2002 Coll. [L. 4]) must be respected during termination of operation.

The presented draft concept for safe termination of operation of a nuclear power plant is the first draft solution for such problems in the process of construction that shall be further developed and refined in the following stages. It is based on current knowledge about technologies and procedures, and on currently valid legal regulations. During the operation of NPP, the technical equipment will be certainly developed and the experience from the decommissioning of similar nuclear installations will be gradually obtained. It will be possible to apply new knowledge on the next levels of the licensing documents and their updates, performed in accordance with the legal regulations.

Certain aspects providing safe termination of operation and decommissioning of NPP must be considered already from the start of the project preparation process of construction, when the technological procedures, equipment, materials and process layout have to be planned in order to facilitate the whole process after the final reactor shutdown.

In connection with the increasing number of implemented as well as ongoing projects of NPP decommissioning in the world, the requirements are applied and the technical as well as organisational measures leading to safe and efficient decommissioning, acquired on the basis of practical experience, are implemented already in the design phase of new nuclear facilities.

The design of new nuclear facilities shall reflect the principles leading to facilitation of future decommissioning:

- Reduction in the number of components and the volume of building material to be disposed of during decommissioning,
- Limitation of the possibility of contamination propagation due to leaks:
    - Reduction in the number of built-in pipe ducts in floors and walls
    - Reduction in the use of underground tanks, sumps and drains for radioactive substances
    - Separation of process systems using radioactive and nonradioactive medium
    - Use of linings on embedded pipelines for ease of maintenance
    - Preference for direct pipelines to limit the formation of deposits
- Selection of suitable material composition for structural parts of the reactor - materials that are directly exposed to neutron flux or are in contact with primary medium. For example, the use of alloys with low content of Co and Ni will lead both to the reduction of activation products 60Co and 63Ni in the material itself and to the reduction of their propagation in the primary circuit.
- Application of suitable chemical regimes that will stabilise corrosion layers and prevent corrosion products from spreading in the primary circuit (e.g. by selecting pH in the range from 6.9 to 7.4, adding corrosion inhibitors in the primary medium),
- Use of modular cast building segments with steel sheathing and use of fixing elements/reinforcements only to the necessary extent (such measures should allow for easy decontaminability, prevent concrete contamination in the case of leaks and degradation mechanisms at metal/concrete interface),
- Reduction in the use of hazardous materials (e.g. PVC, mineral fibres),
- Surface finishing that will allow for easy decontamination and prevent potential contamination from penetrating (application of epoxy varnishes, steel sheathing)
- Selection of technologies not leading to accumulation of hazardous and radioactive materials (activation of liquid and solid radioactive waste treatment technologies)
- Selection of technologies facilitating the access to contaminated systems and their dismantling
- Feasibility of replacement of heavy components (steam generator, pumps) – in terms of logistics and manoeuvrability
- Feasibility of removal or segmentation of large components (pressure vessel, in-core parts)
- Viability of future management of the radioactive waste from decommissioning (processing, storage, transport, disposal)
- If necessary, decontamination by means of remote handling

- Establishment of documentation archiving and operational data collection system for the needs of decommissioning (e.g. archival of metal structural samples, samples of used building materials). New digital 3D technologies allow for the design and recording of systems and components already in the design phase, taking decommissioning into account
- Environmental data archiving for the purposes of environmental impact assessment

Therefore, observance of such principles will mainly ensure minimisation of hazardous, toxic and radioactive waste, maintenance of the effective dose rate for personnel and the population based on the ALARA principle as well as adequate environmental protection.

The following description of the concept has been developed with regard to the requirements of future documents required for issue of licenses for the individual activities pursuant to Act No. 18/1997 Coll. [L. 2]. According to the legal regulations, these documents will have to demonstrate the safety of all activities that are mainly associated with decontamination, dismantling, radioactive waste management, nuclear safety and radiation protection, physical protection and emergency preparedness (emergency plan).

5.2 INITIAL DATA

The important initial source of infromation for the preparation of Chapter 5 of the Initial Safety Analysis Report are the following data and facts known about the construction and operation of currently operated Temelín NPP Units 1 and 2 and information about the projects of new units (generation III and III+ pressurized water reactors) as well as applicable legal regulations and international recommendations, in particular:

- Decree No. 185/2003 Coll., on decommissioning of nuclear installation or category III or IV workplace [L. 260], Decree No. 307/2002 Coll. [L. 4],
- Publicly available documents of suppliers of new nuclear installation,
- ČEZ's principles for NPP decommissioning adopted by the State Office for Nuclear Safety,
- IAEA recommendations and requirements:
    - WS-R-5 - Decommissioning of Facilities Using Radioactive Material, 2006 [L. 261]
    - IAEA SSR-2/1. Safety of Nuclear Power Plants: Design, 2012 [L. 252]
    - IAEA-TECDOC-1657 - Design Lessons Drawn from Decommissioning of Nuclear Facilities; 2011 [L. 263]

In the next stages of documentation preparation, the following background documents, prepared by the supplier of the new nuclear installation (ETE3,4), will be used in this area:

- „Pre-feasibility study of Decommissioning“
- „Feasibility study of Decommissioning“

5.3 PREPARATION FOR DECOMMISSIONING

5.3.1 MEASURES TO BE IMPLEMENTED IN THE OPERATION PERIOD OF ETE3,4

The date of the definitive NPP shutdown will have to be decided sufficiently ahead of the schedule - approximately 5 years is expected - before the start of the stage of termination of operation. The reasons include the following activities and measures that have to be performed and implemented before the start of this stage:

- Draft update of decommissioning plans during operation (every 5 years during operation), development and proposal of decommissioning options
- Preparation of technical documents for the elaboration of decommissioning plans (collection of existing documents, inventory of active and inactive systems, radiation situation mapping, etc.)
- Preparation of documents for a licence for decommissioning (safety analyses, environmental impact assessment documentation, etc.)
- Preparation of projects for the stage of termination of operation (detailed design of decommissioning activities – including all-professional staffing)
- Technical-organisational and managerial activities (administrative procedures, supplier selection, etc.)

During the period of preparation for decommissioning it will be necessary to consider and evaluate the experience from the operational period and incorporate it in the preparatory and implementation project documentation. That applies mainly to the data from the following areas:

- Maintenance and operation
- Evaluation of the results of regular overhauls
- Modernisation and modification of systems and components
- Results of evaluation of the radiation situation during the operation

5.4 DRAFT CONCEPT FOR SAFE TERMINATION OF OPERATION

5.4.1 DESCRIPTION OF MAIN ACTIVITIES AND PROCEDURES DURING TERMINATION OF OPERATION

The termination of operation is the first stage of decommissioning when the reactor is shut down and the spent fuel is removed to the storage pool. In terms of time it is defined as the period when the spent fuel remains in the pools for the required time.

The main activities expected in this stage are as follows:

- Termination of operation and inspection of the state (operation limit, stabilisation of systems, evaluation of system reuse)
- Storage of spent nuclear fuel in NPP and its continuous removal for storage in the Spent Nuclear Fuel Storage Facility
- Drainage and drying of all non-operating systems
- Sampling of shut down, drained and dried systems in order to specify the radioactivity inventory after the termination of the reactor operation
- Removal of liquids from the systems
- Primary circuit decontamination in order to reduce dose rates
- Processing and treatment of waste from decontamination
- Disposal of hazardous materials and wastes
- Processing and treatment of spent ion-exchange resins
- Processing and treatment of other operating wastes
- Ionising radiation monitoring by radiation control system (shall be based on the approved monitoring programme for radiation control in normal operation and shall be implemented using the existing means of radiation control, innovated as necessary)
- Preparation of the programme of staff radiation protection against ionising radiation for the next stage
- Provision of physical protection to the premises of NPP (in the same extent as in normal operation of NPP)
- Provision of emergency preparedness (modification and implementation of the emergency preparedness plan for NPP conditions in decommissioning process)
- Isolation of further operated equipment
- Sale/removal of systems and components or, if appropriate, remaining inventory for further licensing (contaminated, uncontaminated)
- Provision of basic equipment and materials for the needs of decommissioning activities

All activities related to dismantling and demolitions of unnecessary equipment and structures outside the "nuclear island" will also be carried out in the stage of termination of operation. The activities will be carried out as needed and in accordance with the operator's work schedule with regard to the use of work means and workforce of NPP.

For this period all premises relations that were provided during the operation of power plant should be provided:

- Underground services (pipeline, cable, transport, telecommunication, etc.)
- Water supply (drinking, fire, service, demineralized, etc.)
- Provision of electric power
- Supply of heat, cool, heating steam and pressure air
- Storage of chemicals and preparation of solutions
- Collection, treatment, inspection and discharge of waste water
- Spent fuel storage
- Technology for processing and treatment of radioactive waste
- Administrative structures of the premises
- In the structures directly connected to the operation of NPP units, the systems for the reception, reloading and storage of spent fuel including the auxiliary cleaning systems, systems of special air-conditioning including the ventilation stack, radiation control, systems for collection and treatment of waste water, storage of liquid and solid radioactive waste, decontamination system, radiochemical control, etc., will be in operation.
- Physical protection system

Activities carried out during termination of operation will take place in terms of ensuring the proper level of nuclear safety, radiation protection, emergency preparedness, and physical protection under the conditions on the level of normal operation of NPP and will not increase risk in these areas in any case.

5.4.2 RADIOACTIVE WASTE FROM DECOMMISSIONING

Radioactive waste management during termination of operation and decommissioning of NPP shall be in accordance with the legal regulations of the Czech Republic and shall meet the requirements laid down by international documents relating to this area.

All liquid and solid waste from operation including radioactive, hazardous and, where appropriate, toxic materials are expected to be removed and the radiation situation monitored before the termination of NPP operation.

Basic characteristics of the stage of termination of operation in terms of radioactive waste inventory and processing:

- Waste from operation shall be processed and stored in the radioactive waste repository before the start of this stage
- Water from spent nuclear pool shall be treated after the removal of spent nuclear fuel to the Spent Nuclear Fuel Repository

- Unnecessary operating liquids shall be gradually removed and processed
- Water from decontamination of the primary circuit shall be treated
- Dismantling and demolitions in the structures of "nuclear island" will not be carried out

Radioactive waste production in this period will be given by waste of operating nature, i.e. waste water, used filter media, solid waste, HVAC filter elements as well as specialized radioactive waste, which includes liquid radioactive waste from decontamination of the primary circuit and spent fuel pool water.

Radioactive waste is expected to be processed and treated in the same manner as radioactive waste from the period of NPP operation and shall be processed on the existing equipment. The fact that in the decommissioning period the radioactive waste processing and treatment technologies could be on higher level than they are currently shall be taken into account in the decommissioning documentation during regular updates.

Waste, treated to the final product, shall be transported to the radioactive waste repository in the same manner as in the period of NPP operation (road, rail transport).

5.4.3 RADIATION SITUATION MONITORING

In the period before the termination of operation, radiation monitoring shall be carried out as in normal operation of NPP. The system shall monitor the work environment, persons, technology process, discharges and the vicinity of NPP. In this period of time, the requirements for radiation monitoring shall be increased due to the provision of initial data on radiation situation for the needs of termination of operation or, if appropriate, decommissioning.

The radiation monitoring concept and solution shall be based on the concept adopted for NPP and shall also be used during termination of NPP operation. The surface contamination of work surfaces, equipment and means of transport shall be controlled by portable instruments and analysis of the samples taken.

5.4.4 EXPECTED DATE OF TERMINATION OF OPERATION

According to current assumptions, the start of trial operation of the first unit of new nuclear installation (Unit 3) has been set for 2024. Based on this assumption and the requirement for the extension of the life of the NPP, the operation of this unit, with 60-year operation of NPP, will be terminated in approximately 2084. To start decommissioning of the second unit of new nuclear installation (Unit 4) in relation to the start of its operation, the date against the first unit is expected to be shifted by approximately 2 years.

5.5 EXPECTED DECOMMISSIONING CONCEPT

Based on characteristic activities carried out during the individual stages of decommissioning, two basic decommissioning strategies could be proposed:

- Immediate decommissioning, when the decommissioning activities will follow in a single material and time stage
- Deferred decommissioning, when the decommissioning activities are divided into several successive material and time-defined stages with a possible deferral

For illustration, the below diagrams show the schedule of both decommissioning methods.

5.6 IMMEDIATE DECOMMISSIONING

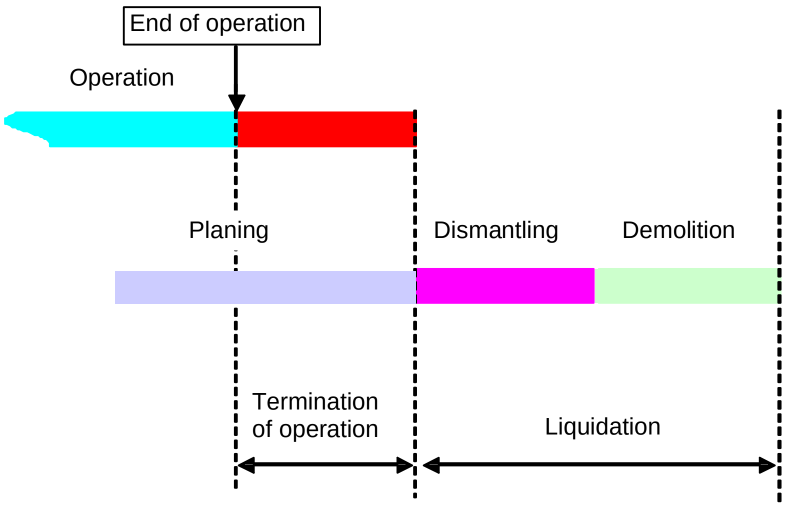

Fig. 66 Immediate decommissioning diagram

5.6.1 OPTION 1: IMMEDIATE DECOMMISSIONING

As far as this option is concerned, decommissioning activities will be carried out smoothly and continuously from the moment of the start of decommissioning (permanent reactor shutdown) to the termination of decommissioning (release of equipment for the use for different purposes). The process is divided into the following stages:

- Termination of operation
- Preparation for liquidation
- Liquidation

5.7 DEFERRED DECOMMISSIONING

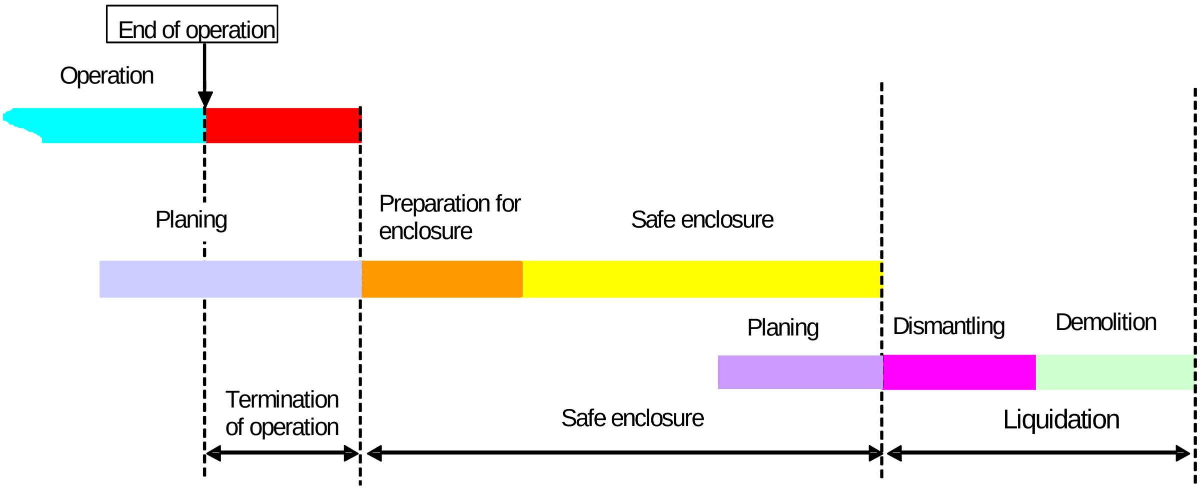

Fig. 67 Deferred decommissioning diagram

Three decommissioning options have been selected for the already operated units of NPP Temelín. An immediate decommissioning option and two deferred decommissioning options, when a different range of safe enclosure has been defined. In one option, the minimum extent of safe enclosure is defined with regard to the layout of the main generating unit – taking into account the minimum necessary arrangements to provide safety barriers to undismantled systems, components and structures; in the second option, the extent of enclosure is extended to cover all structures where contaminated materials are found. The process is divided into stages that can be characterized by the activities to be carried out - decontamination, dismantling, demolition, radioactive waste processing and treatment, disposal of inactive waste, etc.

5.7.1 OPTION 2: DEFERRED DECOMMISSIONING - REACTOR SAFE ENCLOSURE

A basic feature of this option is the creation of the safe enclosure of reactors inside the containment while maintaining the foundation part of the reactor building. Activated and contaminated process equipment as well as civil structures will be located inside for the period of time planned to reduce the activity of radionuclides by their natural decay. This option can be characterized by an interrupted flow of the decommissioning process, with the meantime of the safe enclosure and then its removal until the release of equipment for different purposes. Two separate structures (containments with reactors and foundation parts) will be left on site for the specified safe enclosure period.

The process is divided into the following stages:

- Termination of operation
- Preparation for the safe enclosure of reactors
- Safe enclosure of reactors
- Removal of the safe enclosure of reactors

5.7.2 OPTION 3: DEFERRED DECOMMISSIONING - SAFE ENCLOSURE OF ACTIVE FACILITIES

A basic feature of Option 3 is the enclosure of all facilities of the "nuclear island" after the termination of operation of the reactor units and after the facilities are put into a safe condition for the safe enclosure. Activated and contaminated process equipment as well as civil structures are located inside the facilities for the period of time planned to reduce the activity of radionuclides by their natural decay. Neither dismantling of active facilities nor post-dismantling decontamination is carried out in the preparatory period of the safe enclosure. All necessary repairs and maintenance of civil structures (civil barriers) are performed. The facilities are enclosed for the determined period of time, during which contamination reduces as a result of radionuclide decay. After the determined enclosure period elapses, the facilities will be removed. This option can be characterized by an interrupted flow of the decommissioning process, with the meantime of the enclosure of active facilities followed by removal of the safe enclosure (performance of other decommissioning activities – deferred decommissioning) until the release of equipment for the use for different purposes.

The process is divided into the following stages:

- Termination of operation
- Preparation for the safe enclosure of active facilities
- Safe enclosure of active facilities
- Removal of the safe enclosure of active facilities

In terms of the decommissioning strategy at the Temelín site, the assumption that this site will probably continue to be used for the commercial purposes of ČEZ, a. s., is important. This assumption follows from the concept of managing the land and structures used for the generation of electric power in the Czech Republic and from the requirement for maximum funds saving in constructing the new power facilities. Therefore it will be appropriate to choose the method of classification of individual structures at the site into the groups as needed for decommissioning of the nuclear installation. These are four groups of facilities - active civil structures ("nuclear island" facilities), inactive civil structures auxiliary (facilities operated for the needs of the decommissioning process), essential inactive civil structures (facilities difficult to dismantle and demolish) and other civil structures (nonessential facilities with the possibility of decommissioning already in the period of preparation for removal). According to the current concept and objectives of the energy sector in the Czech Republic, the dismantling procedure could be selected depending on the needs of the considered plan of the use of Temelín NPP site.

The immediate decommissioning option is currently considered for decommissioning of ETE3,4. A detailed schedule and description of activities shall be developed in the next phase of documentation preparation in accordance with the above-mentioned procedures. The documentation shall contain a description of removal of active and inactive systems and components.

6 ASSESSMENT OF QUALITY ASSURANCE IN THE PROCESS OF SELECTION OF SITE, METHOD OF QUALITY ASSURANCE FOR PREPARATORY STAGE OF CONSTRUCTION AND QUALITY ASSURANCE PRINCIPLES FOR LINKING STAGES

Work carried out during siting of the new nuclear installation ETE3,4 was divided into four basic groups by linkage and relation to the relevant legal regulations of the Czech Republic. The work on the new nuclear installation was divided in accordance with the following regulations:

- Pursuant to Act No. 100/2001 Coll., on Environmental Impact Assessment [L. 255]
- Pursuant to Act No. 18/1997 Coll., on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act) and on Amendments and Additions to Related Acts [L. 2] as well as pursuant to the implementing legal decrees (decrees of the State Office for Nuclear Safety), in particular Decree No. 215/1997 Coll. [L. 1]
- Pursuant to Act No. 183/2006 Coll., on Town and Country Planning and the Building Code (Building Act) [L. 256]
- Pursuant to Act No. 137/2006 Coll., on Public Procurement [L. 257]

The subject matter of this part of the Initial Safety Analysis Report is to assess the quality assurance system in performing work and activities associated with the selection of the site, specifically work and activities primarily focused on meeting of the requirements and conditions defined by Act No. 18/1997 Coll. [L. 2] and the implementing decrees of the State Office for Nuclear Safety, especially Decree No. 132/2008 Coll. [L. 258].

The legal regulations for the nuclear area were and are followed in ensuring and performing such activities. The documentation for the issue of a licence for siting of a nuclear installation was prepared to meet all requirements laid down by this legislation.

In addition, the requirements and provisions of other legal regulations, standards and other codes related to the problems dealt with in the individual parts of the documentation were taken into account in creating the documentation. The scope of application is always specified in the text of the relevant part of the documentation.

6.1 APPLICANT'S QUALITY SYSTEM

ČEZ, a. s. assures quality in compliance with the established quality management system. A management system, which defines the basic management areas, is applied in ČEZ, a. s. following the rules [P. 1]. These basic areas are further divided into the individual management areas and individual processes. Each such area and process has its own guarantor with delegated powers and responsibilities. The management system is described in the rules [P. 2] and the requirements for the area of the management system are laid down in the standard [P. 3].

The Quality Management System (QMS) for nuclear energy is based on the guideline [P. 4] and on the standard [P. 5] as well as on the requirements defined in the applicable legal regulations and IAEA recommendations for the area of quality management. On the basis of the New Vision project, ČEZ, a. s. is obliged to continuously improve the established quality management system.

QMS is established in accordance with the requirements arising from the applicable legal regulations, mainly Act No. 18/1997 Coll. [L. 2], as well as Decree No. 132/2008 Coll. [L. 258]. The established QMS ensures that the requirements laid down in Sections 3 to 14 of Decree No. 132/2008 Coll. [L. 258] are met, i.e. that the corresponding requirements are assigned to all basic management areas and processes, and that they are met. These provisions of the cited regulation relate to suppliers to the appropriate extent.

The current control documentation is available in an electronic form on the Intranet of ČEZ, a. s., in the ECM controlled documents application.

The fundamental principles for the management and organisation system in ČEZ Group are defined following the rules [P. 6].

The principles for the management and organisation system in the Investment Division are defined in the rules [P. 7].

The mission, competencies and activities of the individual departments within the VJE Department in the Investment Division are defined in the CEO's order [P. 8] and in the document [P. 9].

The organisational structure of the VJE Department forms a part of the document [P. 10]. The electronic organisational structure is available on the Intranet of ČEZ, a. s.

The Project Safety Management is carried out in compliance with the principles [P. 1], as described in the rules [P. 11]. The rules [P. 11] define the basic competencies and responsibilities.

There are three levels of safety management defined in ČEZ Group that are compatible with the levels in line management:

- Group,
- Segment,
- Executive.

A level of the Segment Management is applied to the Project, which is controlled by the Safety Management Segment Centre. Based on the general binding legal regulations and selected international recommendations, the Safety Management Segment Centre defines safety requirements.

Each part of the segment independently incorporates the requirements in its documentation and ensures activities in compliance with such requirements.

A safety manager is appointed for the Project in the segment level - Quality and Safety Manager, who cooperates on the creation of safety requirements and oversees their fulfilment.

The quality assurance for the licensed activities is carried out by means of the established QMS. The description of the established system is presented to the State Office for Nuclear Safety in the form of Quality Assurance Programmes (QAP). These documents are prepared in accordance with the method [P. 12].

The quality requirements, imposed on suppliers, are defined in the contract. The relevant supplier elaborates these requirements in Quality Plans and in Project Manuals that are approved by the relevant departments of ČEZ, a. s.

The compliance with the requirements pursuant to Act No. 18/1997 Coll. [L. 2], is secured and controlled in accordance with the method [P. 13].

In ČEZ, a. s., the supplier selection and procurement management is described in the guideline [P. 14], operating procedures [P. 15], [P. 16], [P. 17] and methods [P. 18], [P. 19] and [P. 20].

In ČEZ, a. s., the establishment of QMS and compliance with the requirements defined in Decree No. 132/2008 Coll. [L. 258] are verified by means of internal (in ČEZ, a. s.) and external (suppliers) audits following the procedure [P. 21]. The audits are carried out in accordance with an annual audit plan, compiled by the Quality Management Department. For verification of suppliers for the stage of siting see the document [P. 10].

Internal audits are carried out in accordance with the method [P. 22] and procedure [P. 23]. External audits at suppliers are carried out in accordance with the procedure [P. 24]. The outputs of the internal and external audits are stored in an electronic form in the information system of ČEZ, a. s.

Independent supplier inspections are also carried out, under the concluded contracts of work and the approved quality plans. The objective of customer audits of the suppliers of ČEZ, a. s., is to systematically verify professional competence and qualification of the existing as well as potential suppliers in compliance with the customer requirements, legal regulations, harmonized technical standards and codes.

QMS is assessed on a periodic basis for the period of a calendar year and the results are summarized in an assessment report that is presented to the management for consideration, which defines the necessary measures, if needed.

6.2 ASSESSMENT OF QUALITY ASSURANCE OF SELECTION OF SITE

The **Investor** (owner), applicant for licence under Section 9(1) a) of Act No. 18/1997 Coll. [L. 2], and customer is:

Tab. 164 Investor's identification data

Business Name	ČEZ, a. s.
Organisation Form	joint-stock company
Identification Number	45274649
Registered Office	Duhová 2/1444, 140 53 Prague 4, Czech Republic
Licensee's Place of Business	Headquarters ČEZ, a. s., Duhová 2/1444, 140 53 Prague 4, Czech Republic Temelín Nuclear Power Plant, Temelín, Postcode 373 05
Scope of Business	Electricity production, production of heat, etc. (see the Commercial Register). Note ČEZ, a. s., will be the Client, i.e. Investor, for the construction of electricity and heat production facility.

Identification of direct suppliers:

Tab. 165 Suppliers' identification data

Business Name	**Ústav jaderného výzkumu Řež a.s.**
Organisation Form	joint-stock company
Identification Number	46356088
Registered Office	Husinec-Řež, No. 130, Postcode 250 68
Business Name	**EMPRESARIOS AGRUPADOS INTERNACIONAL S.A.**
Organisation Form	joint-stock company
Identification Number	Madrid Company Register, Book of Joint-Stock Companies, Volume 4567, Folio 34, Page M-75212
Registered Office	Magallanes 3, 28015 Madrid, Spain
Business Name	**Euroenergy, spol. s r.o.**
Organisation Form	limited liability company
Identification Number	45797340
Registered Office	Švédská 1538/22, Prague 5, Postcode 150 00
Business Name	**GEFOS a.s.**
Organisation Form	joint-stock company
Identification Number	25684213
Registered Office	Kundratka 17, Prague 8, Libeň, Postcode 180 82
Business Name	**SCES – Group, spol. s r.o.**
Organisation Form	limited liability company

Identification Number	48288268
Registered Office	Petrská 1178, Prague 1, Nové Město, Postcode 110 00
Business Name	**ARTECH spol. s r.o.**
Organisation Form	limited liability company
Identification Number	25024671
Registered Office	Stroupežnického 1370, Ústí nad Labem, Postcode 400 01
Business Name	**ENVIROS, s.r.o.**
Organisation Form	limited liability company
Identification Number	61503240
Registered Office	Na Rovnosti 1, Prague 3, Postcode 130 00

A licensing document [P. 10] "Quality Assurance Programme for Siting of Nuclear Installation 'Units 3 and 4' at Temelín Site" was prepared for the purposes of quality assurance under Section 13(5) of Act No. 18/1997 Coll. [L. 2]. This document was prepared by ČEZ , a. s., in 04/2010. The document was prepared in compliance with the requirements defined in Decree No. 132/2008 Coll. [L. 258], and was approved by Decision No. SÚJB/OTBIS/16140/2010 issued by the State Office for Nuclear Safety on 29 June 2010. The document is updated in accordance with the requirements on a regular basis.

Documentation relating to the quality, listed in the below table, was prepared by suppliers to ensure supplier quality. This documentation was reviewed and approved by ČEZ, a. s.

ÚJV Řež a.s.

Tab. 166 Documentation ÚJV Řež, a.s.

Document	Document title
PLJ 2500.14	Quality Plan – Preparation of the documentation for Site Decision proceedings for the "Completion of Temelín NPP" project in the scope for the preparation of the Project Plan, including engineering and design support of the Client in associated activities
PLJ 500.09	Quality Plan – New nuclear unit preparatory phase

Other suppliers

Tab. 167 Documentation of other suppliers

Supplier	Document identification	Document title
GEFOS a.s.	PISŘ	Integrated Management System Manual
Empresarios Agrupados Intarnacional S.A.	00-MC-X-00001-/I	Quality and Environmental Management Manual
Empresarios Agrupados Intarnacional S.A.	Manual BIS4112008	Project Manual: Documentation for the selection of supplier (s) of new nuclear power plant (bid invitation specification)
Euroenergy	Manual BIS4112008	Project Manual: Documentation for the selection of supplier (s) of new nuclear power plant (bid invitation specification)

Supplier	Document identification	Document title
SCES – GROUP	P01	Integrated Quality and Environmental System Manual
ARTECH	P01	Integrated Quality, Environmental and OHS System Manual
ENVIROS	PS	Quality Management Manual

The principal processes important to the stage of siting in terms of nuclear safety and radiation protection were defined in the QAP. The evaluation of these processes is included in Section 6.2.1

Throughout the siting stage, the quality was assured in accordance with the established quality management system of ČEZ, a. s., as described in the approved QAP [P. 10]. The suppliers carried out the activities associated with the siting stage in accordance with this QAP, quality requirements arising from the contract of work and the presented quality plans.

6.2.1 EVALUATION OF PROCESSES WITH IMPACT ON NUCLEAR SAFETY AND RADIATION PROTECTION

The processes that are important in terms of nuclear safety and radiation protection, which are critical for obtaining a licence for siting, were defined in the approved QAP [P. 10]. These processes were divided as follows:

- Processes and activities under licensee's responsibility (internal processes – IP),
- Processes and activities carried out by suppliers (supplier processes – SP).

6.2.1.1 EVALUATION OF INTERNAL PROCESSES (IP)

All internal processes were assured in accordance with the rules defined in the documents [P. 1], guideline [P. 4] and in QAP [P. 10]. The processes were primarily assured by ČEZ, a. s., specialists, specifically from the VJE Department within the Investment Division in cooperation with the Production Division.

6.2.1.1.1 PV1 – Setting of Fundamental Principles for Project Management (In the Spirit of the PDCA Cycle, i.e. Planning, Implementation, Feedback, Improvement)

The objective of the PV1 process was to set the fundamental principles for Project management throughout the Project, especially by means of the method of PDCA cycle.

Setting of the fundamental principles for Project management is based on the established quality management system and on the rules [P. 1].

The siting process followed and shall continue to follow the guideline [P. 25], procedure [P. 26] and the documentation [P. 9], which was issued directly for the Completion of ETE3,4 Project.

For responsibilities and powers of the individual departments cooperating on the Project see the CEO's Order [P. 8].

Setting of the fundamental principles for Project management is described in [P. 9], where the method for Project planning, management, implementation, monitoring, control and closing as well as the method for motivating all participants to achieve the objective of the Project are defined.

PV1 process management was evaluated on the inspection day of the Project, when the individual phases of the Project are discussed. Meeting minutes are stored in the information system of ČEZ, a. s.

6.2.1.1.2 PV2 – Provision/Preparation, Review and Approval of Internal Documentation of ČEZ, a. s., In the Phase of Opportunity Management (Feasibility Study, Business Plan, Project Plan)

The objective of the PV2 process was to ensure review, verification and approval of internal documents – Feasibility Study, Business Plan and Project Plan, which are aimed at assessing the Project from the technical, economic, environmental, social and legal point of view.

The preparation of the Feasibility Study was assigned to suppliers based on a tender.

Suppliers' capacities to meet the requirements for the work including quality assurance requirements in accordance with Decree No. 132/2008 Coll. [L. 258] were reviewed and verified within the tender.

The tender for the preparation of the necessary documents was terminated by signing contracts with the selected suppliers:

- EMPRESARIOS AGRUPADOS INTERNACIONAL, S. A,
- Ústav jaderného výzkumu Řež a.s.,
- Euroenergy, spol. s r.o.

During the preparation of the Feasibility Study, ČEZ, a. s., specialists monitored the progress and preparation in the form of consultations and inspection days. The outputs of consultations and inspection days were recorded and their incorporation in the Feasibility Study being created was subsequently controlled. All records created during the preparation are archived.

The Feasibility Study was received from contracted suppliers in 09/2007.

In the Feasibility Study, the Project was evaluated from the technical, economic, environmental, social and legal point of view. The Feasibility Study was prepared in accordance with the contract requirements in the specified quality and in accordance with the applicable Project manual. ČEZ, a. s., personnel prepared the internal document - Business Plan based on the data provided in the Feasibility Study. The prepared Business Plan was commented on by internal specialists of ČEZ, a. s., and then submitted to the management of ČEZ, a. s., for approval, in 05/2009. All records made during this process are stored with the project manager responsible for the preparation of the Business Plan. Based on these internal documents, a decision on the continuation of the Project was made.

The process is currently still running and will be terminated by preparing and approving the Project Plan by the management of ČEZ, a. s.

6.2.1.1.3 PV3 – Siting Process – Additional Survey, Additional Evaluation and Additional Confirmation of Site (Provision of Support Studies, Provision of Additional Surveys, Provision of Analyses and Provision/Preparation of Initial Safety Analysis Report)

The objective of the PV3 process was to ensure verification and summarisation of the results of the suitability assessment of site with regard to intended siting of the new nuclear installation ETE3,4. The verification included the assessment of topicality of the already prepared surveys, studies and analyses for ETE1,2 and the Spent Nuclear Fuel Repository, and the preparation of new surveys, studies and analyses. The results of site assessment are well-arranged in Chapter 2 of this Initial Safety Analysis Report.

The siting process follows the method [P. 13].

Surveys, studies and analyses, prepared during siting of ETE1,2 and during siting of the Spent Nuclear Fuel Repository, were used for the process of siting of ETE3,4, and additional surveys, support studies and analyses were further performed (see PD2 evaluation). The prepared "Feasibility Study", which included the verification of the possible technical solutions of ETE3,4, was used as a basic interface for the verification of already prepared surveys, studies and analyses, and for the identification of the need for new surveys, studies and analyses

New surveys, studies and analyses including the preparation of Chapters 2 and 4 of this Initial Safety Analysis Report were assigned to suppliers based on a tender and under the contract concluded with Ústav jaderného výzkumu Řež, a.s. The results of surveys, studies and analyses were continuously assessed and verified while preparing the Initial Safety Analysis Report in cooperation with the specialists from Ústav jaderného výzkumu Řež a.s.

Chapters 1, 3, 5 and 6 of this Initial Safety Report were prepared by ČEZ, a. s, specialists. During the preparation, the progress and preparation were continuously monitored in the form of consultations and inspection days. The outputs of consultations and inspection days were documented in the form of the relevant records and incorporated in the document being created. Technical solutions were checked and verified for correctness by internal professional specialists. The incorporation of all requirements arising from consultations and inspection days was checked and verified before final submission.

Records obtained from the progress of this process are stored with the manager of the partial project Procedure under the Atomic Act in the Site Decision Proceedings Stage. For Project Organisation and the list of partial projects see the document [P. 9].

The process management resulted in the preparation of the Initial Safety Analysis Report in the required quality in accordance with the contract requirements and applicable internal documents. The results of site assessment for siting of the new nuclear installation ETE3,4 are well-summarized in Chapter 2 of this Initial Safety Analysis Report.

6.2.1.1.4 PV4 – Provision of Documentation for Other Licensing Procedures – EIA Process, EURATOM TREATY Notification, Site Decision Pursuant to the Building Act

The objective of the PV4 process was to ensure preparation of the environmental impact assessment (EIA) documentation for the construction of ETE3,4. An application for the Site Decision shall be submitted based on the issued EIA decision and a licence issued by the State Office for Nuclear Safety under Section 9(1) a). The objective of the notification according to the EURATOM TREATY will be to notify the European Commission and the Member States of the European Union of planned construction of a nuclear installation and of radioactive discharges and radioactive waste management.

The process of preparation of the EIA documentation followed the applicable method [P. 27] and Act No. 100/2001 Coll. [L. 255]. The process of preparation of the EIA documentation included mainly the following activities:

- Notification under Section 6 of Act No. 100/2001 Coll., submitted in 07/2008
- Fact-finding procedure under Section 7, carried out from 08/2008 to 02/2009 under ref. No. 8063/ENV/09
- EIA documentation under Section 8, prepared and subsequently submitted to the Ministry of the Environment in 05/2010
- Expert report under Section 9, issued in 02/2012
- Public hearing under Sections 9 and 17 – in 06/2012
- Transboundary assessment under Sections 11, 12 and 13, throughout the EIA process
- Statement under Section 10 is expected by the end of 2012

The process of preparation of the EIA documentation and elaboration of support studies was monitored by the licence applicant's specialists in cooperation with contracted suppliers listed below:

- SCES – Group, spol. s. r. o.
- Ústav jaderného výzkumu Řež, a. s.
- ARTECH spol. s. r. o.
- ENVIROS, s. r. o.

The EIA documentation was prepared by contracted suppliers in the specified quality in accordance with the contract requirements, Act No. 100/2001 Coll. [L. 255] and internal documents. The EIA documentation was considered in compliance with the mentioned act. Records obtained from the progress of this process are stored with the manager of the partial project EIA Process - Transboundary.

Article 41 of the EURATOM TREATY notifies the European Commission of planned construction of a nuclear installation. The notification documentation shall be submitted to the European Commission before signing a contract with the selected supplier. During the notification process, the European Commission may request additional information. The notification process will be terminated by issuing a statement of the European Commission on Article 41 of the EURATOM TREATY.

The licence applicant (Investor) shall be responsible for the preparation of the documentation.

Article 37 of the EURATOM TREATY notifies the Member States of the European Union of radioactive discharges and management of radioactive waste from the newly constructed nuclear installation. The licence applicant (Investor) shall be responsible for the preparation of the documentation. This documentation shall then be submitted to the State Office for Nuclear Safety that is responsible for its submission to the Member States. The expected date for documentation submission is before the issue of a licence for operation of a nuclear installation or, if appropriate, before the issue of a licence for construction.

The activities within this process are continuing and will be terminated by:

- Issuing the Site Decision for construction (i.e. after the issue of the decision of the State Office for Nuclear Safety on a licence under Section 9(1) a) of Act No. 18/1997 Coll. [L. 2])
- Terminating the notification process according to Articles 37 and 41 of the EUROATOM TREATY

6.2.1.1.5 PV5 – Procurement; Preparation of Tender Documents, Enquiry, Bid Evaluation, Supplier Selection, Conclusion of Contracts, Control and Supervision of Suppliers During Work Implementation

The objective of the PV5 process was the procurement of tender documents and their sending to the potential suppliers in the form of an enquiry.

The entire process of supplier selection for the Project follows Act No. 137/2006 Coll. [L. 257] and internal documents issued by ČEZ, a. s.

The data for the preparation of tender documents was processed under contracts concluded with suppliers:

- EMPRESARIOS AGRUPADOS INTERNACIONAL, S. A.
- Ústav jaderného výzkumu Řež, a. s.
- Euroenergy, spol. s. r. o.

Capacities of the above listed suppliers to meet the quality requirements defined in Decree No. 132/2008 Coll. [L. 258] were reviewed, verified and presented in the document [P. 10].

In accordance with the approved Project manual, ČEZ, a. s., monitored compliance with the quality requirements during the fulfilment of the contract. In the manual, relations among the cooperating entities were set and inspection mechanisms were defined in the form of inspection days and consultations. Records obtained from inspection days and consultations are archived.

Internal specialists from ČEZ, a. s., commented on the submitted data for tender documents in the secured software Caliber. The data was processed in accordance with the contract requirements and Project manual. Complete tender documents were submitted to potential suppliers in the form of a bid in Autumn 2011.

The contract concluded with EMPRESARIOS AGRUPADOS INTERNACIONAL, S.A. included the requirement for the preparation of the Bid Evaluation Manual. The prepared manual was received and modified to its final form by ČEZ, a. s., specialists. In accordance with this manual, any bid submitted by suppliers on the basis of an enquiry shall be evaluated.

Bid evaluation will result in the selection of a supplier, with whom the contract for the supply of Temelín NPP Units 3 and 4 will be concluded.

Under the concluded contract, the progress of performance shall be monitored by the Procurement Department in cooperation with the NPP Construction Department throughout the duration of the contract. The NPP Construction Department shall also be responsible for compliance with the requirements for customer audits at suppliers, for items that require supervision of supply quality. To carry out supervision of suppliers, inspection mechanisms shall be defined in the contract and the principle of awareness of all participants shall be applied. For requirements for control and supervision see the document [P. 17].

6.2.1.1.6 PV6 – Project Documentation and Record Management

The PV6 process ensured Project documentation and record management to make sure that any document and record is at any time traceable, legible and any changes recorded.

Documentation (including record) management, creation, issue and further life cycle are described in the rules [P. 2]. For the conditions and requirements for document management see the documents [P. 28] and [P. 29].

ETE3,4 Project documentation is administered by the Control, Documentation and Information Systems Department. Basic rules are described in the documents [P. 9] and [P. 30].

All employees are informed about the documents by means of an application in the company's information system, where a demonstrable familiarisation is ensured at the same time. All managers shall be responsible for demonstrable familiarisation with the documentation.

In ČEZ, a. s., the principles for documenting the activities in the form of records are defined. As the records differ considerably for the individual processes, a rule is set that the relevant process guarantor lays down the requirements and tools for their identification, storage, protection, retrieval and retention period. Record management meets the following requirements:

- Records are legible throughout their life and easy to identify
- Records are easy to retrieve and traceable throughout their life
- Records cannot be changed (possible changes can be made only in the defined controlled manner)

For a list of records see the particular control document in the Section "Documentation Outputs".

The method of keeping and storage of documents and records created in the ETE3,4 siting stage was controlled in accordance with the requirements arising from the internal documentation [P. 30].

6.2.1.1.7 PV7 – Management of Non-Conformities, Remedial and Precautionary Measures against Project Non-Conformities

The PV7 process - management of non-conformities, remedial and precautionary measures - is aimed at identifying non-conformities, making sure they are corrected and defining such measure to prevent their reoccurrence to the maximum possible extent.

The process of management of non-conformities, their remedy and precautionary measures against non-conformities is set in accordance with the requirements defined in Section 8 of Decree No. 132/2008 Coll. [L. 258]. The mechanisms for identification and handling of non-conformities are established in the procedure [P. 31] that is binding upon the entire ČEZ, a. s. company.

The process of handling of a non-conformity includes its identification, detection of its root cause, derivation of remedial measure, identification of any associated non-conformities and derivation of precautionary measures. In the case of major non-conformity, a comprehensive review of the efficiency of the implemented measures is carried out as a part of closing the handling of non-conformity.

Having regard to the nature of processes/activities in this phase of the Project, a simplified system of handling of non-conformities is used within the NPP Construction Department, which includes the following:

- Identification of a non-conformity
- Detection of its cause
- Remedy (elimination of non-conformity)
- Specification of remedial measures to prevent the non-conformity from reoccurring
- Specification of precautionary measures to prevent potential non-conformities from occurring
- Verification of implementation and efficiency of remedy, remedial and precautionary measures

Non-conformity records are kept in the information system of the NPP Construction Department. The Quality and Safety Manager is responsible for administration.

In the case of identification of a non-conformity, the established remedial or precautionary measures shall be continuously monitored and evaluated in terms of the state of their introduction and efficiency on Project inspection day.

6.2.1.1.8 PV8 – Assessment of Established Quality System for Licensed Siting Activity In Terms of Its Efficiency and Effectiveness

The PV8 process ensured assessment of the established quality system in terms of its efficiency and effectiveness.

The basic rules for assessment are described in the document [P. 40].

The Project quality system was assessed by the Quality and Safety Manager in cooperation with CEO and NPP Construction Managers.

This assessment is documented in an assessment report that is prepared at the beginning of the next calendar year.

The assessment report SMQ2010 was prepared and approved in 03/2011.

The assessment report SMQ2011 was prepared and approved in 04/2012.

The established quality system was assessed for every period of the calendar year in accordance with the requirements defined in the internal documents. This assessment results in the above mentioned assessment reports.

6.2.1.1.9 PV9 – Project Human Resource Management

The PV9 process made sure that the persons working on the Project in the siting stage have the necessary knowledge, skills and experience.

The qualification requirements for human resources are laid down in the guideline [P. 32]. Based on this document, the qualification requirements for workers of the NPP Construction Department were laid down, and obtainment and maintenance of necessary knowledge, skills and experience were ensured.

The process of ensuring employee qualification included, but was not limited to, preparation and implementation of special training, professional training, tests, and examination of medical or mental fitness with a view to creating conditions for safe performance of work.

All records of education, training, experience and compliance with the qualification requirements are kept in a uniform manner for the entire ČEZ, a. s. company.

The process of human resource planning and management for the Project in the siting stage was carried out in compliance with the internal documents. An organisational structure of the NPP Construction Department was specified for the siting stage, which qualitatively and quantitatively reflected the current situation in the Project.

6.2.1.2 EVALUATION OF SUPPLY PROCESSES (SP)

The quality assurance in supply processes followed:

- Concluded contracts of work,
- Quality assurance programme for siting of the nuclear installation "Unit 3 and 4" at Temelín NPP,
- Supplier quality plans,
- Manuals and documentation related to the established supplier quality management system.

6.2.1.2.1 PD1 – Preparation of the Feasibility Study

The objective of the PD1 process was to prepare documentation aimed at assessing the Project from the technical, economic, environmental, social and legal point of view The Feasibility Study for ČEZ, a. s., was prepared under contracts concluded with suppliers in relation to fulfilment of the PV2 process (Section 6.2.1.1.2 of this Initial Safety Analysis Report).

The "Project Manual: Feasibility Study of the New Nuclear Power Plant in the CR" was developed for the Feasibility Study project. This manual was binding upon all participating parties throughout the project. This manual defines the relations among cooperating entities including inspection mechanisms.

All data for the Feasibility Study was obtained and collected from surveys, analyses and applicable legal regulations. All data was verified and evaluated for its suitability for ETE3,4 before its use.

During the preparation, the progress and preparation were continuously controlled in the form of consultations and inspection days. A final consultation took place at the conclusion. The outputs of consultations were documented in the form of the relevant records and incorporated in the document being created. The incorporation of all requirements arising from consultations was checked and verified before final submission.

The specification was prepared and submitted in 09/2007.

In the Feasibility Study, the Project was evaluated from the technical, economic, environmental, social and legal point of view. The Feasibility Study was prepared in accordance with the contract requirements in the specified quality and in accordance with the applicable Project manual.

6.2.1.2.2 PD2 – Siting Process – Additional Survey, Additional Evaluation and Additional Confirmation of Site (Provision of Support Studies, Provision of Additional Surveys, Provision of Analyses and Provision/Preparation of Initial Safety Analysis Report)

The objective of the PD2 process was to verify and summarise of the results of the suitability assessment of site with regard to intended siting of the new nuclear installation ETE3,4. The verification included the assessment of topicality of the already prepared surveys, studies and analyses for ETE1,2 and the Spent Nuclear Fuel Repository, and the preparation of new surveys, studies and analyses. The results of assessment were then summarized in the prepared Initial Safety Analysis Report.

ÚJV Řež a.s. Divize Energoprojekt prepared, under the contract of work, Chapters 2 and 4 of this Initial Safety Analysis Report including Analysis of Needs and Means of Providing Physical Protection.

Data obtained from surveys and analyses of the Temelín construction site performed approximately over the last 30 years were available for the assessment of the site for ETE3,4. During the preparation of the Initial Safety Analysis Report, the original data was verified under Section 6 of Decree No. 215/1997 Coll. [L. 1], to make sure that it did not become worthless. The updated data and surveys were checked and verified by professional specialists from ÚJV Řež a.s. Divize Energoprojekt. Records of this verification are kept. New surveys and analyses were assigned in the case of non-current data or new methods of preparation.

Siting criteria for a nuclear installation in accordance with Decree No. 215/1997 Coll. [L. 1] and site requirements in accordance with the standard IAEA NS-R-3 [L. 6] were used to assess the suitability of the site including taking account of the new safety guide issued by the State Office for Nuclear Safety BN-JB-1.14, Interpretation of Criteria for the Siting of Nuclear Installations and a Proposal for Evidence of Compliance [L. 268].

In addition, other IAEA standards relating to siting of nuclear installations were used for the assessment of the site.

Section 2.1 of this Initial Safety Analysis Report summarizing basic data of site (geography and demography) was prepared using the method of special compilation of site related facts on the basis of collection of information from public information servers and from created reports and analyses.

Sections 2.2 and 2.3 of this Initial Safety Analysis Report dealing with external and internal events were prepared on the basis of analyses identifying sources of events in accordance with the IAEA recommendation NS-G-3.1 [L. 9]. After the identification of sources of risk, the analysis continued by identifying and two-phase screening the events. Preliminary screening was carried out either by using the screening distance value (SDV) or by evaluating the probability of the occurrence of the event. Events which could not be ruled out by screening were assessed with respect to their frequency and the effects of the interaction of the events with the new nuclear installation.

To prepare Sections 2.2 and 2.3 of this Initial Safety Analysis Report new analyses relating to human effects (external and internal) and updated data concerning air traffic in the vicinity of Temelín NPP were prepared, the impact of maximum design basis accident of a long-distance gas pipeline on facilities of the new nuclear installation was assessed and the electromagnetic environment was assessed. The sources of internal risks were identified only on the premises of ETE1,2 and the sources of risks for ETE3,4 will be available once the design has been prepared.

To evaluate the meteorological conditions for the Temelín site (Section 2.4 of this Initial Safety Analysis Report) the data obtained by measuring and observing at stations of the Czech Hydrometeorological Institute that is archived in the Clidata digital database was used. Time series and stations with a 30-year measurement period were preferred. A repeated data revision was also performed within the processing of the data. The initial data and the description of evaluation method were elaborated in a report prepared by the Czech Hydrometeorological Institute.

Data for the evaluation of hydrological conditions (Section 2.5 of this Initial Safety Analysis Report) were collected by means of a standard measurement in the network water measuring stations with a continuous level recording and conversion to flow rate through current stage-discharge curves. So-called discharge inventories were prepared for the purposes of providing data beyond the observed profiles. The updated hydrological data was processed by the Czech Hydrometeorological Institute.

To prepare Section 2.5 of this Initial Safety Analysis Report the studies for water draw-off from the Hněvkovice water reservoir and for waste water removal to Kořensko were carried out. Furthermore, the possibilities of danger by flood conditions, extreme precipitations and special floods were assessed. The planned Vltava waterway was also assessed.

In addition to the Preoperational Safety Report for ETE1,2 [L. 18], the data for Section 2.6 of this Initial Safety Analysis Report - Geological, Geotechnical and Seismic Conditions was obtained from official specific map documents issued by state organisational units or by organizations appointed by state (e.g. official topographic or geological maps, maps of undermined territories, protected mineral estates, etc.), as well as from the results of surveys and assessments carried out by different organisations and for different purposes than siting of a nuclear installation (e.g. engineering-geological surveys for different projects on site, results of mineral, geological, hydrogeological surveys, etc.). To keep the data up-to-date, specialized surveys and assessments of the selected site characteristics and risk analyses were carried out both for events and phenomena affecting the safety of a nuclear installation and for the events and phenomena affected by operation of a nuclear installation. These data were sorted and verified.

The additional surveys included, but were not limited to, the following surveying activities – geophysical measurement, geological mapping, and drill-hole survey. Newly implemented research projects on the Paleoseismologic Investigation of Fault Structures in the Vicinity of Temelín NPP and on the Verification of Movement Activity of N-S Faults at Temelín NPP were used during processing.

To evaluate the data a number of methods were used, some of which were already used for the evaluation in the Preoperational Safety Report for ETE1,2 [L. 18] and others were new. To assess seismic danger a new SL-2 calculation method was developed using a probabilistic approach, which shall be followed in revising seismic danger in the next preparatory stages of the new nuclear installation. The engineering-geological risks were assessed using methods extended by new methods recommended by IAEA guide. New paleoseismic methods and procedures were applied to the assessment of risks associated with movements on faults, especially for the purposes of verifying the assumptions of the time of last movements on faults and possibly adding data on prehistoric earthquakes in a seismologic databank. A wide range of research and survey methods were used during field survey of a tectonic history of faults, such as analysis of geological maps, geomorphological analysis of site, evaluation of geophysical properties, geological mapping, evaluation of the occurrence of river terraces, evaluation of archival bore-hole data, bore-hole survey, digging of exploratory trenches, dating of rock samples, analysis of microearthquake occurrence, etc. These procedures are described in detail in research reports and documents.

The radiation situation on site (Section 2.7 of this Initial Safety Analysis Report) was assessed based on a systematic monitoring and a periodic assessment, as described in the relevant methodology of ČEZ, a. s. and in an annual report "Results of Monitoring of Discharges and Radiation Situation in the Vicinity of Temelín Nuclear Power Plant". The method of transport and distribution of radionuclides in the environment was used for the assessment.

The assessment of the site in terms of emergency preparedness in Section 2.8 of this Initial Safety Analysis Report took into account safety objectives defined in the EUR document [L. 264]. In accordance with those safety objectives, the ETE3,4 units shall be designed so that in accident conditions no urgent protective measures need to be introduced at distances greater than 800 m and no follow-up protective measures need to be introduced at distances greater than 3 km.

A detailed elaboration of description and evidence of suitability of the site from the aspect of siting criteria for nuclear installations is included in Chapter 2 of this Initial Safety Analysis Report.

During the preparation, the progress and preparation were continuously controlled in the form of consultations and inspection days. The outputs of consultations and inspection days were documented in the form of the relevant records and incorporated in the document being created. Technical solutions were checked and verified for correctness by internal professional specialists. The incorporation of all requirements arising from consultations and inspection days was checked and verified before final submission.

Records obtained from consultations and inspection days are stored with the manager of the partial project Procedure under the Atomic Act in the Site Decision Proceedings Stage.

The Initial Safety Analysis Report, support studies, additional surveys and analyses were prepared in the specified quality in accordance with the contract requirements and in accordance with the relevant internal documents. The results of site assessment for siting of the new nuclear installation ETE3,4 are well-summarized in Chapter 2 of this Initial Safety Analysis Report.

6.2.1.2.3 PD3 – Preparation of EIA Documentation, Documentation Required for Notification According to the EURATOM TREATY, Documentation for Site Decision

The objective of the PD3 process was to prepare the environmental impact assessment (EIA) documentation for the construction of ETE3,4.

Under the concluded contract, SCES – Group, spol. s r.o., was responsible for a complete preparation of the EIA documentation

The updated data, results of surveys and studies prepared for ETE1,2 and the Spent Nuclear Fuel Repository were used for the preparation of the EIA documentation. All used studies, analyses and surveys are listed in an annex to the EIA documentation.

During documentation preparation, the progress and preparation were continuously controlled in the form of consultations and inspection days. The outputs of consultations and inspection days were documented in the form of the relevant records and incorporated in the document being created. Documentation creation was checked and verified for correctness by internal professional specialists. The documentation was reviewed and the incorporation of all requirements arising from consultations and inspection days was verified before final submission. Records obtained from consultations and inspection days are stored with the manager of the partial project EIA Process – Transboundary.

- The EIA documentation was prepared and submitted to ČEZ, a. s. The EIA documentation was prepared in the required quality in accordance with the contract requirements, Act No. 100/2001 Coll. [L. 255] and internal documents.

6.2.1.2.4 PD4 – Data Processing for Tender Documents

The objective of the PD4 process was to define the requirements for tender documents. Based on tender documents, bids were received that will be evaluated and a contract will be concluded with the selected supplier.

The contract for processing of data for tender documents was concluded with three suppliers: EMPRESARIOS AGRUPADOS INTERNACIONAL, S.A., Ústav jaderného výzkumu Řež, a. s. and Euroenergy, spol. s. r. o.

A project manual was created under the contracts concluded with the above listed suppliers. This manual was binding throughout the processing of data for tender documents, "Project Manual: Documentation for the Selection of Supplier(s) of New Nuclear Power Plant (Bid Invitation Specification)".

The full specification of requirements for tender documents was prepared under the concluded contract and the Project Manual and was subsequently commented on by internal specialists from ČEZ, a. s., in the secured software Caliber.

Every step of documentation preparation was assessed and verified by specialists from ČEZ, a. s., and by suppliers, in terms of correctness, completeness and comprehensiveness. The comments were considered and incorporated in the next version of the documentation. Records of all meetings were kept.

Tender documents were prepared in the specified quality in accordance with the contract specification and as such submitted to potential suppliers of ETE3,4.

6.3 SUMMARY

The processes mentioned in Section 6.2 resulted in the creation of:

- A Feasibility Study that evaluated the Project from the technical, economic, environmental, social and legal point of view. The Feasibility Study was prepared in the required quality under concluded contracts and in accordance with the Project Manual. Based on the data obtained from the Feasibility Study, a business plan was developed, which was approved by the management of ČEZ, a. s.
- EIA documentation, which was prepared in the required quality in accordance with the contract requirements, Act No. 100/2001 Coll. [L. 255] and internal documents [P. 27]. The EIA documentation was considered in accordance with the above mentioned act.
- Tender documents, prepared in the required quality in accordance with the contract requirements and valid Project Manual. Such documents were submitted to potential suppliers for the preparation of their bids.

All relevant quality requirements arising from Act No. 18/1997 Coll. [L. 2], Decree No. 132/2008 Coll. [L. 258] and the IAEA standard NS-R-3 [L. 6] were met during the licensed siting activity.

Chapter 6 of the Initial Safety Analysis Report meets the requirements defined in Act No. 18/1997 Coll. [L. 2], Appendix A, Part I., Subparagraph 5. The preparation respected the contract requirements for quality and the internal documents [P. 13].

6.4 QUALITY ASSURANCE FOR THE PREPARATORY STAGE OF CONSTRUCTION AND QUALITY ASSURANCE PRINCIPLES FOR LINKING STAGES

6.4.1 DESIGN

In accordance with Act No. 18/1997 Coll. [L. 2] and the applicable decrees issued by the State Office for Nuclear Safety, the licensed activity Siting of ETE3,4 is followed by the activity Design of New Nuclear Installation, which is defined by the following milestones:

- The start is the effective date of a Licence issued by the State Office for activity under Section 9(1) a) of Act No. 18/1997 Coll. [L. 2]
- The end is the effective date of a Licence issued by the State Office for activity under Section 9(1) b) of Act No. 18/1997 Coll. [L. 2]

The quality assurance for Design of ETE3,4 shall respect the established quality management system and shall be documented in the relevant Quality Assurance Programme for Design. This programme shall be prepared in accordance with the method [P. 12].

The Quality Assurance Programme for Design shall be submitted to the State Office for Nuclear Safety for approval as soon as possible after signing a contract with the selected supplier.

The procurement activities in the design phase shall be limited to procurement of equipment and services not related to the nuclear part of the plant.

Listed are the principal activities related to designing of ETE3,4 that shall be included in the Quality Assurance Programme for Design:

1	**Preparation, Verification, Approval of the Project (Supplier Documentation)**
	Objective: Well-prepared project
2	**Preparation and Approval of the Preliminary Safety Report and the Proposed Method of Providing Physical Protection**
	Objective: Making sure that the solution defined by the Project meets the requirements for nuclear safety, radiation protection and emergency preparedness laid down by implementing regulations
3	**Preparation and Provision of Documentation for the Individual Licensing Procedures**
	Objective: Provide appropriate data for the relevant licensing procedure
4	**Observance of Set Basic Principles for Project Management**
	Objective: Well-managed Project
5	**Project Documentation and Record Management**
	Objective: Up-to-date documentation provided in a timely manner and at the designated place

6	**Management of Non-Conformities, Remedial and Precautionary Measures against Project Non-Conformities** Objective: Handling of non-conformities, learning from them, prevention of reoccurrence, continuous improvement
7	**Human Resource Management** Objective: Prepared human resources
8	**Assessment of Established Quality System for Licensed Activity** Objective: Feedback, continuous improvement
9	**Inspection and Supervision of Suppliers During Design** Objective: Supplies implemented in a top-quality and safe manner
10	**Safety Management** Objective: Project managed in a safe manner

Ad 1)

Design inputs, such as design basis, legislative requirements, codes and standards and contract specifications shall be reviewed and approved by an organisation responsible for designing. Design methods and software used for designing shall be identified and verified. The draft design shall be verified by persons or an organisation different than the organisation responsible for designing. The draft design shall be verified by qualified employees or a qualified organisation. Methods, tests and software used for the verification of the draft design shall be identified and verified. A list of the selected equipment shall be drawn up in accordance with Decree No. 132/2008 Coll. [L. 258]. Any design changes shall be identified, verified, approved and documented throughout the design period. Interfaces among all participating organisations shall be identified and controlled during the design process. All documents created during the design process shall be stored and archived.

Ad 2)

The preparation of the Preliminary Safety Report shall demonstrate that the proposed design solution meets the requirements for nuclear safety, radiation protection, emergency preparedness, physical protection, fire protection, technical safety, occupational health and safety protection, and environmental protection in accordance with the applicable implementing regulations. A proposed method of providing physical protection shall be prepared. The documentation shall be prepared in accordance with the requirements laid down in Act No. 18/1997 Coll. [L. 2].

Ad 3)

Documents prepared for the individual licensing procedures shall meet all legal requirements, shall be based on a design basis and shall be up-to-date. Such documents shall be stored and archived.

Ad 4)

External and internal interfaces, which shall be uniquely identified, defined and documented, shall be established among organisations cooperating on the Project. All legislative requirements, contract specifications and applicable codes and standards shall be met during the Project.

Ad 5)

Preparation, issue and changes of control documents shall be documented and before its introduction, this document shall be subjected to review in terms of appropriateness, adequacy, efficiency, comprehensibility, correctness and completeness, and approved by persons appointed for this purpose. Documents and records shall be comprehensible, complete, uniquely identifiable, traceable and available in their full version at any time. Documents shall be protected against damage, loss or theft. All documents shall be stored and archived for a specified period of time. A demonstrable familiarisation of employees shall be ensured. Requirements and responsibilities for document and record management shall be laid down.

Ad 6)

A documented procedure shall be created for non-conformity management that shall lay down the requirements for identification of non-conformity, the non-conformity reporting procedure, method of handling of a non-conforming item and the requirements for preventing misuse of the non-conforming item, including the plan for removal of non-conformity, and the requirements for the evaluation of consequences of the non-conformity. Non-conformities shall be remedied in accordance with the documented procedure for non-conformity management.

A documented procedure shall be prepared and introduced to eliminate potential non-conformities. Any introduced remedial or precautionary measures shall be regularly monitored and evaluated in terms of the state of their introduction and efficiency.

Ad 7)

During the Project, an appropriate organisational structure shall be created on the part of ČEZ, a. s., with the adequate responsibilities and powers. The qualification requirements for employees shall be laid down, and obtainment and maintenance of necessary knowledge, skills and experience shall be ensured. Fulfilment of the qualification requirements shall be documented.

Ad 8)

A plan for quality system assessment (internal audits, supplier assessment, self-assessment) shall be created. Plans for system assessment shall contain the scope, requirements, assessment team, activities to be assessed, used documents and documented procedures to be used as a basis for system assessment. A final report shall be issued after every assessment. All records obtained from system assessment shall be stored and archived.

Ad 9)

The procurement activities in the design phase shall be limited to procurement of equipment and services not related to the nuclear part of the plant.

Requirements imposed on suppliers and subcontractors shall be laid down in contract documents. Contract documents shall specify the scope of supply, technical requirements, quality assurance requirements, documentation requirements, and software access rights.

Ad10)

Safety requirements shall always be the top priority and it shall be ensured that other requirements (economic, environmental, qualitative, protection, etc.) will not be assessed separately from safety requirements to eliminate their possible adverse impact on safety.

Compliance with safety requirements during the Project shall be systematically controlled. Legal requirements, requirements laid down by the licence applicant, data obtained from safety analyses, previous operational experience, research results and proven technical procedures shall be used for safety assessment. Project safety shall be assessed by individuals or a group different from the group elaborating the Project.

Project safety shall be assured by special qualified and trained personnel on all levels. Safety culture shall be maintained on all organisational levels. During the Project, the organisation shall continuously develop and improve the safety culture.

A Quality Assurance Programme for Design shall be prepared by ČEZ, a. s., as a principal quality system document in order to demonstrate provision of activities during Design of ETE3,4, specifically for:

- Activities carried out by a licensee
- Activities carried out by supply entities

The Quality Assurance Programme for Design shall be submitted to the State Office for Nuclear Safety for approval by the applicant, ČEZ, a. s., so as to be approved before the commencement of design activities, as required by Section 13(5) of Act No. 18/1997 Coll. [L. 2].

The quality assurance system for activities performed by supply entities shall be defined, in this licensed activity, in Quality Plans prepared by supply entities that shall be approved by the licensee, ČEZ, a. s. The Quality Assurance Programme as well as suppliers' Quality Plans for Design shall, for items, processes and activities included therein:

- Stipulate the method of quality assurance in the scope of Sections 3 to 14 of Decree No. 132/2008 Coll. [L. 258]
- Relate to formerly prepared Quality Assurance Programmes and Quality Plans in order to maintain continuity in quality assurance for ETE3,4
- Respect and observe decisions issued by the State Office for Nuclear Safety within the licensing procedure under Section 9(1) a) of Act No. 18/1997 Coll. [L. 2], which shall be related to the licensed activity Design of ETE3,4

6.4.2 CONSTRUCTION

The licensed activity Design of ETE3,4 is followed by the licensed activity Construction of ETE3,4 – licensed under Section 9(1) b) of Act No. 18/1997 Coll. [L. 2].

The method of quality assurance and provision of activities during Construction shall be performed in accordance with the established Quality Management System and shall be documented in the Quality Assurance Programme for Construction. This programme shall be prepared in accordance with the method [P. 12].

The Quality Assurance Programme for Construction shall be submitted to the State Office for Nuclear Safety for approval so as to be valid at least one year before submitting an application for a licence for construction.

Activities to be dealt with in the Quality Assurance Programme for Construction include, but are not limited to:

1	**Procurement of Equipment and Services (Procurement)**
	Objective: Equipment and services procured in a top-quality and safe manner in accordance with specifications
2	**Preparation of Construction Site including Site Facilities**
	Objective: Construction site and site facilities prepared in accordance with design specifications
3	**Civil Work**
	Objective: Civil work carried out in accordance with design specifications
4	**Technology Manufacture and Installation**
	Objective: Manufacture and installation carried out in a top-quality and safe manner
5	**Transport, Handling and Storage of Technology on Construction Site**
	Objective: Technology provided in a timely and safe manner and at the designated place
6	**Inspections and Tests, Identification of State after Inspections and Tests**
	Objective: Inspections and tests performed in accordance with specifications
7	**Identification and Control of Items, Traceability**
	Objective: All items are identified and traced
8	**Metrology**
	Objective: Devices are calibrated and have the specified accuracy
9	**Preparation, Verification and Approval of Documentation Created During Construction**

	Objective: Well-prepared documentation
10	**Provision of Supervision on the Construction Site (Nuclear Safety, Radiation Protection, Emergency Preparedness, Fire Protection, and Occupational Health and Safety Protection)**
	Objective: Safe construction site
11	**Preparation and Approval of the Preliminary Safety Report and the Proposed Method of Providing Physical Protection**
	Objective: Making sure that the solution defined by the Project meets the requirements for nuclear safety, radiation protection and emergency preparedness laid down by implementing regulations
12	**Provision of Documentation for the Individual Licensing Procedures**
	Objective: Provide appropriate data for the relevant licensing procedure
13	**Observance of Set Basic Principles for Project Management**
	Objective: Well-managed Project
14	**Project Documentation and Record Management**
	Objective: Up-to-date documentation provided in a timely manner and at the designated place
15	**Management of Non-Conformities, Remedial and Precautionary Measures against Project Non-Conformities**
	Objective: Handling of non-conformities, learning from them, prevention of reoccurrence, continuous improvement
16	**Human Resource Management**
	Objective: Prepared human resources
17	**Assessment of Established Quality System for Licensed Activity**
	Objective: Feedback, continuous improvement
18	**Safety Management**
	Objective: Project managed in a safe manner

Ad 1)

Specific requirements for the procurement of equipment and services shall be included in contract documents. Activities carried out by suppliers shall be supervised (inspection and test plan, inspection days) and assessed (audits, supplier assessment). Compliance with specific requirements shall be declared in supplier documents for procured equipment and service. A documented procedure for handling of non-conformities shall be prepared. All records made during procurement of equipment and services shall be stored and archived. The above mentioned requirements apply to the entire supply chain.

Ad 2)

The construction site shall be prepared in compliance with conditions of a licence, relevant legal requirements, technical regulations, technical conditions, contractually binding conditions and supplier documentation. Preparation of construction site and site facilities shall be documented to ensure compliance of outputs with design requirements.

Ad 3)

Civil work shall be carried out in compliance with conditions of a licence, relevant legal requirements, technical regulations, technical conditions, contractually binding conditions and supplier documentation. The specification shall lay down requirements for special processes (e.g. welding, forming, non-destructive testing, heat treatment). Special processes shall only be carried out by persons holding the necessary qualification, which shall be verified. The progress of construction shall be documented to ensure compliance of outputs with design requirements. All documents and records shall be stored and archived.

Ad 4)

Manufacture and installation shall be carried out in compliance with conditions of a licence, relevant legal requirements, technical regulations, technical conditions, contractually binding conditions and supplier documentation. The specification shall lay down requirements for special processes (e.g. welding, forming, non-destructive testing, heat treatment). Special processes shall only be carried out by persons holding the necessary qualification, which shall be verified. The progress of manufacture and installation shall be documented to ensure compliance of outputs with design requirements. All documents and records shall be stored and archived.

Ad 5)

Transport, handling and storage of equipment shall be carried out in such a manner to prevent damage, destruction, unauthorized use or loss. A procedure shall be declared in specific cases. This procedure shall be in compliance with legal and safety regulations.

Ad 6)

Quality assurance requirements and acceptance criteria shall be stipulated in specifications related to ordered equipment and services. Inspections and tests shall be performed in accordance with documented procedures, instructions, lists, drawings and other adequate means. In accordance with the specification, inspection points shall be defined for equipment and services. The state of inspections and tests shall be provided directly on an item or in a document related to an item. The state shall be provided by means of indicators such as location of an item, identification, inspection records or in another adequate manner. Persons performing inspections and tests, and inspectors, shall hold the necessary qualification. Inspection authorities shall participate pursuant to Act No. 18/1997 Coll. [L. 2], and other legal decrees. Records demonstrating quality assurance shall be stored and archived.

Ad 7)

Inspection of items shall make sure that conforming items shall be used and installed. Items for manufacture shall be identified from receipt of an order, through manufacture up to use or installation. The selected and unselected equipment shall be identified using such materials and methods to ensure that the identification shall be clear, legible and unique throughout the life of an item without affecting it. Identification of an item shall be consistent in documents that shall be supplied with this item.

Ad 8)

Tools, instruments and other measuring and testing equipment shall be controlled, calibrated within the specified periods of time and checked for accuracy in accordance with the specified limits. Devices and equipment shall be adequately identified. Records of such activities shall be stored.

Ad 9)

Documentation prepared during construction shall be reviewed, verified and approved by a licensee and its suppliers. Documentation shall be up-to-date, complete, unique, comprehensible and uniquely identifiable so as to be available at any time. Documentation shall be protected against damage, loss or theft. Documentation shall be stored and archived.

Ad 10)

Supervision shall be provided on the construction site during all phases - preparation of the construction site, construction, manufacture, storage, installation, and performance of inspections and tests in order to meet the requirements for nuclear safety, radiation protection, emergency preparedness, fire protection and occupational health and safety protection in accordance with applicable legal regulations.

Ad 11)

The preparation of the Preliminary Safety Report shall demonstrate that the proposed design solution meets the requirements for nuclear safety, radiation protection, emergency preparedness, physical protection, fire protection, technical safety, occupational health and safety protection, and environmental protection in accordance with the applicable implementing regulations. A proposed method of providing physical protection shall be prepared. The documentation shall be prepared in accordance with the requirements laid down in Act No. 18/1997 Coll. [L. 2].

Ad 12)

Documents prepared for the individual licensing procedures shall meet all legal requirements, shall be based on a design basis and shall be up-to-date. Such documents shall be stored and archived.

Ad 13)

External and internal interfaces, which shall be uniquely identified, defined and documented, shall be established among organisations cooperating on the Project. All legal requirements, technical specifications and applicable codes and standards shall be met during the Project.

Ad 14)

Preparation, issue and changes of control documents shall be documented and before its introduction, this document shall be subjected to review in terms of appropriateness, adequacy, efficiency, comprehensibility, correctness and completeness, and approved by persons appointed for this purpose. Documents shall be protected against damage, loss or theft. All documents shall be stored and archived for a specified period of time. A demonstrable familiarisation of employees shall be ensured.

Ad 15)

A documented procedure shall be created for non-conformity management that shall lay down the requirements for identification of non-conformity, the non-conformity reporting procedure, method of handling of a non-conforming item and the requirements for preventing misuse of the non-conforming item, including the plan for removal of non-conformity, and the requirements for the evaluation of consequences of the non-conformity. Non-conformities shall be remedied in accordance with the documented procedure for non-conformity management.

A documented procedure shall be prepared and introduced to eliminate potential non-conformities. Any introduced remedial or precautionary measures shall be regularly monitored and evaluated in terms of the state of their introduction and efficiency.

Ad 16)

During the Project, an appropriate organisational structure shall be created on the part of ČEZ, a. s., with the adequate responsibilities and powers. The qualification requirements for employees shall be laid down, and obtainment and maintenance of necessary knowledge, skills and experience shall be ensured. Fulfilment of the qualification requirements shall be documented.

Ad 17)

A plan for quality system assessment (internal audits, supplier assessment, self-assessment) shall be created. Plans for system assessment shall contain the scope, requirements, assessment team, activities to be assessed, used documents and documented procedures to be used as a basis for system assessment. A final report shall be issued after every assessment. All records obtained from system assessment shall be stored and archived.

Ad 18)

Safety requirements shall always be the top priority and it shall be ensured that other requirements (economic, environmental, qualitative, protection, etc.) will not be assessed separately from safety requirements to eliminate their possible adverse impact on safety.

Compliance with safety requirements during the Project shall be systematically controlled. Legal requirements, requirements laid down by the licence applicant, data obtained from safety analyses, previous operational experience, research results and proven technical procedures shall be used for safety assessment. Project safety shall be assessed by individuals or a group different from the group elaborating the Project.

Project safety shall be assured by special qualified and trained personnel on all levels. Safety culture shall be maintained on all organisational levels. During the Project, the organisation shall continuously develop and improve the safety culture.

The Quality Assurance Programme for Construction shall be submitted to the State Office for Nuclear Safety for approval by the applicant, ČEZ, a. s., within the licensing procedure for construction so as to be approved before the issue of a licence for construction by the State Office for Nuclear Safety under Section 9(1) b) of Act No. 18/1997 Coll. [L. 2], as arising from Section 13(5).

The quality assurance system for activities performed by supply entities shall be defined, in these stages, in Quality Plans prepared by supply entities that shall be approved by the licensee, ČEZ, a. s. The Quality Assurance Programme as well as suppliers' Quality Plans for Construction shall, for items, processes and activities included therein:

- Stipulate the method of quality assurance in the scope of Sections 3 to 14 of Decree No. 132/2008 Coll. [L. 258]
- Relate to formerly prepared Quality Assurance Programmes and Quality Plans in order to maintain continuity in quality assurance for ETE3,4
- Respect and observe decisions issued by the State Office for Nuclear Safety for previous licensed activities

6.4.3 LINKING STAGES

The stage of Construction of ETE3,4 shall be followed by licensed activities under Section 9(1) of Act No. 18/1997 Coll. [L. 2]. These activities include:

- Individual Stages of Commissioning of ETE3,4 – licensed under Section 9(1) c) of Act No. 18/1997 Coll. [L. 2]
- Operation of ETE3,4 - licensed under Section 9(1) d) of Act No. 18/1997 Coll. [L. 2]

The quality assurance system in linking licensed activities shall be carried out in accordance with the established Quality Management System and shall be documented in the relevant Quality Assurance Programme for the relevant activity. This programme shall be prepared in accordance with the method [P. 12].

The Quality Assurance Programmes for linking licensed activities shall be submitted to the State Office for Nuclear Safety sufficiently in advance to allow for their approval before commencement of the licensed activity.

6.4.3.1 INDIVIDUAL STAGES OF COMMISSIONING OF NUCLEAR INSTALLATION

Commissioning of a nuclear installation shall be carried out in compliance with Section 9(1) c) of Act No. 18/1997 Coll. [L. 2], and related Decree No. 106/1998 Coll. [L. 259]. Commissioning is expected to be carried out in the following stages:

- Inactive Testing – tests performed to verify functionality of a nuclear installation without loading nuclear fuel
- Active Testing – tests performed to verify functionality of a nuclear installation with loaded nuclear fuel followed by trial operation

The quality assurance system and the performance of activities in the stage Commissioning of Nuclear Installation shall be carried out in accordance with the established Quality Management System and shall be documented in the Quality Assurance Programme for Individual Stages of Commissioning. This programme shall be prepared in accordance with the method [P. 12].

Any change made to the existing installation Temelín NPP (ETE1,2 and Spent Nuclear Fuel Repository) and its operating regulations, induced by construction and individual stages of commissioning of ETE3,4, shall be handled as changes made to the existing operated nuclear installation. They shall always be handled individually and separately in accordance with the scope of change. This issue shall be specified with the State Office for Nuclear Safety during the licensed activity Construction.

6.4.3.2 OPERATION OF ETE3,4

Quality assurance during operation of ETE3,4 shall follow the current Quality Management System and shall be documented in the Quality Assurance Programme for Operation of ETE3,4 that shall become a part of documentation for quality assurance for all licensed activities. This programme shall be prepared in accordance with the method [P. 12].