Stress tests of nuclear power plants – ČEZ, a.s.

Evaluation of Nuclear Safety and Safety Margins of Temelín NPP

(on the background of events at the Fukushima NPP)

Ref. Number: **ČEZ_ETE_001r00**		Date:	Signature:
RECCOMENDED:	Production Division Safety Director		
	Central Engineering Director		
	Asset Management Director		
SUBMITTED:	Temelín NPP Director		
APROVED:	Production Division Director		

ANOTACE

Stress tests of nuclear power plants required by the European Council are defined as focused assessment of safety margins and resistance of nuclear plants, on the background of events that occurred at the Fukushima-Daiichi nuclear plant in Japan following the tsunami on March 11, 2011. The specification requires to analyze combinations of extreme conditions leading to a severe accident of the nuclear facility disregard of their low probability. This must be taken into account when reading and studying the described events.

Based on the facts concerning the accident in Fukushima-Daichi NPP international nuclear institutions issued number of conclusions and lessons learned for the nuclear industry and national nuclear regulators, applicable to all reactor types. The presented report contains results of stress tests, specified in declarations of ENSREG (European Nuclear Safety Regulators Group) dated 13 March 2011 „EU Stress Tests Specifications“. The stress tests form part of comprehensive safety evaluation of NPPs, referring to the international documents published in relation to the given event, e.g.:

WANO SOER 2011-2, Fukushima Daiichi Nuclear Station Fuel Damage Caused by Earthquake and Tsunami, March 2011

SOER 2011-3, Fukushima Daiichi Nuclear Station Spent Fuel Pool/Pond Loss of Cooling and Makeup, August 2011

INPO Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Station, November 2011

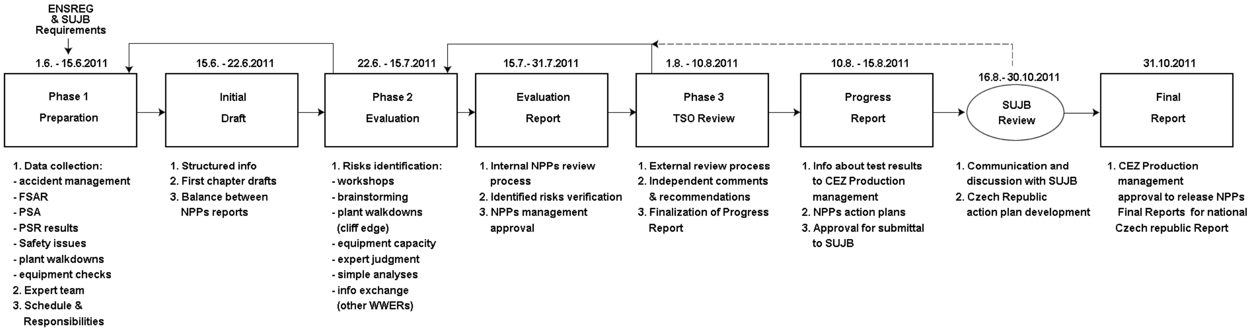
IAEA International fact finding expert mission of the Fukushima-Daichi NPP accident following the great east Japan earthquake and tsunami, 16 June 2011

US NRC Recommendation for enhancing reactor safety in the 21th century, 12 July 2011

ČEZ, a.s. has been requested by SÚJB to perform the stress tests by a letter dated 25 May 2011. Stress tests procedure has been specified by instruction of ČEZ, a.s. production division director, which specified its scope and method.

Evaluation has been conducted by experts in nuclear safety, nuclear facility designing, accident management, emergency preparedness and severe accident phenomenology research, fully qualified for this activity. The evaluators proceeded in accordance with the deterministic approach assuming gradual failure of all preventive measures in evaluation of extreme scenarios.

Considering the short period of time available for stress test performance a work team has been appointed, a fixed schedule has been set and output of individual phases were defined. As of 15 August SÚJB received information on the current state and progress of the evaluation in the form of so called Progress Report.



Several working meetings with stress tests implementers from other VVER type NPPs within so called VVER club (Dukovany, Paks, Loviisa, Bohunice, Mochovce and Kozloduj NPPs) took place during evaluation processing. Discussions with other non EU VVER type NPPs take place within WANO MC.

To ensure objectivity of the evaluation an independent review of the results was provided by recognized external contractors active in nuclear safety, including ÚJV Řež and Westinghouse.

The below text evaluates the characteristics of Temelin NPP and its location based on knowledge resulting from safety assessment analyses, studies, surveys and engineer's estimates and related to the occurrence of unexpected (SAMG) and unlikely conditions with the expected rate of one occurrence being 1 case per 1,000,000 years of NPP operation, or even less.

The assessment was performed by experts in nuclear safety, designing of nuclear facilities, accident management, emergency preparedness and phenomenology research of severe accidents, fully qualified for this activity. The evaluators proceeded in compliance with deterministic approach of expected successive failure of all preventive actions during evaluation of extreme scenarios.

The assessment includes:

- Review of design requirements and their compliance;
- Review of resistance and robustness against beyond design basis conditions (safety margins, diversity, redundancy, physical separation, etc.) and defense-in-depth

system efficacy, including the identification of cliff edge effects and possible actions to avoid them, and

- Identification of all ways and means to maintain 3 fundamental safety functions (reactivity, fuel cooling, leak prevention) as well as support functions (electrical power supply, ultimate heat sink) considering efficient options to further improve the defense-in-depth.

The assessment covers all operating modes and states of plant. It specifically deals with impacts of events such as earthquake, floods, extreme natural effects, loss of off-site power, station blackout, or loss of ultimate heat sink. A significant part of the report is the chapter “Severe Accidents” which describes processes and strategies to manage severe accidents in different stages. The present report evaluates them and describes details beyond the scope of licensing requirements set in the current legislation (Act no. 18/1997 Coll.).

The assessment results corroborate the fact that Temelin NPP robustness provides considerable margins to avert severe accidents. The strengths in terms of external risk primarily include:

- Design subject to continuous control and applying the relevant new knowledge;
- Robust and conservative design ready to manage severe conditions;
- Location with minimum seismic risk;
- Location practically eliminating external floods;
- Two large storage reservoirs to replenish raw water;
- Extensive inventory of cooling water on the site;
- Compact spent nuclear fuel pools ensuring fuel subcriticality even if flooded with clean water; and
- Placement of a spent nuclear fuel pool inside the full pressure containment.

In the past Temelín NPP safety has been confirmed by number of international missions reviewing standard of nuclear safety with regard to experience of western NPPs. In particular the following missions are concerned:

- Site Safety Review, Design Review, NUS Halliburton (1990): assessment of the siting, safety systems, core design and safety analysis,
- Pre–Operational Safety Review (Pre-OSART), IAEA (1990): plant construction and operation preparation review,

- Pre–Operational Safety Review Follow up, IAEA (1992): evaluation of fulfillment of recommendations from 1990,
- Quality Assurance Review IAEA (QARAT) (1993): quality assurance review,
- Leak Before Break Application Review, IAEA (1993 – 1995): applicability of LBB method in accordance with the worldwide practice,
- Fire Safety IAEA (1996): compliance with international trends in fire protection,
- International Peer Review Service – PSA 1, PSA 2, IAEA (IPERS), (1995 – 1996): review of scope of PSA methodology implementation,
- Safety issues of WWER 1000 Resolution Review, IAEA (1996): review of safety issues identified by IAEA for VVER1000/320 reactor types,
- Physical Protection Assurance – IPPAS, IAEA (1998): concept of the physical protection and physical guarding assurance during construction,
- Operational Preparedness and Plant Commissioning Review, IAEA (2000): review of Management, Organization and Control, Operation, Maintenance and Commissioning areas,
- Operational Safety Review (OSART) IAEA (2001): NPP´s operational safety review,
- Safety Issues of WWER 1000 Resolution Review Follow up, IAEA (2001): check of the status of safety issues identified in IAEA-EBP-VVER-05,
- International Physical Protection Advisory Service (IPPAS), IAEA (2002): review of the final state of NPP´s physical protection assurance,
- Site Seismic Hazard Assessment – expert mission, IAEA (2003): follow up of the 1990 mission, review of the results of local seismic monitoring networks in NPP´s region (confirmation of sufficient 0.1 g acceleration value for the seismic class (SL2).
- International Probabilistic Safety Assessment Review, IAEA (IPSART) (2003): review of newly updated PSA models for internal initiation events,
- Operational Safety Review Follow-up (OSART FU), IAEA (2003): review of plant´s response to recommendations from 2001.
- Peer Review, WANO (2004): review of NPP´s operational safety by other nuclear power plant operators,
- Peer Review Follow-up, WANO (2006): review of fulfillment of recommendations in the areas of Organization and Control, Maintenance, Engineering and Chemistry.
- WANO Peer Review (2011): periodical review of management of NPP areas by other nuclear power plant operators

Site assessment is a significant part in view of stress testing. Although the Czech territory is not highly affected (except of floods) by extreme weather phenomena, the official IAEA criteria set in a generally binding regulation since 1979 were fully satisfied while selecting a site for Temelin NPP. This regulation specified siting of future NPPs in view of minimizing risk relating to external effects.

For that reason, the location of Temelin NPP is considered as highly stable in relation to seismicity. The ultimate heat sink is atmosphere and cooling is provided by evaporation from cooling towers or pools with spray systems. Raw water is supplied to NPP from storage reservoirs installed much lower than the NPP ground level, consequently their damage would not jeopardize safety of the power plant (e.g. due to extreme floods, earthquake, etc.).

Safety margin assessment is primarily based on design analyses. Where the results of design analyses cannot determine safety margins, their engineer's judgment is used. The report analyzes a possible impact of combined unavailability of NPP safety related (critical) systems, e.g. in the event of station blackout, loss of systems providing ultimate heat sink, or loss of off-site power (e.g. power grid breakdown).

The performed analyses identified the following main opportunities to increase resistance of Temelin NPP in the preventive as well as the consequent stage of escalation of extreme conditions at nuclear power plants:

- Add more mobile sources of electrical power supply and use of fluids, independent and fully separated from the existing design systems (by power supply, dislocation, etc.)
- Add the organization and staff training for management of extreme conditions (e.g. when both NPP units are affected, or in case of loss of control centers, communication systems, etc.)
- Extend the capacity of hydrogen treatment system to eliminate undesirable consequences of excessive hydrogen production during severe accidents and increase the resistance of NPP Temelin containment.

In 2008 and 2009 complex safety review after 10 years of Temelín NPP operation took place (so called PSR - Periodical Safety Review for all 14 areas in compliance with IAEA NS-G-2.10 instruction). It was an in depth review of fulfillment of requirements of domestic and international legislative documents, WENRA requirements defined in "Reactor Safety Reference Levels" documents as well as other international recommendations of IAEA documents (Safety Guides, INSAG). Comprehensive review within PSR identified similar areas for safety improvement as those included in this report. Some of them (improvement off NPP design resistance to consequences of severe accidents, including capacity increase of hydrogen disposal system by adding recombiners for severe accidents) are already being processed for implementation and they would be implemented even regardless this new evaluation. PSR assumes implementation of most of the measures until 2015 or until next PSR (2018) in justified cases.

Results of the targeted review of safety margins and NPP´s resistance, required by the European Council, confirm efficiency and appropriateness of before adopted decisions to implement measures resulting in improved resistance of the original design. No issue was identified which would require an immediate action. The power plant is capable to manage safely even highly improbable, extreme emergency situations, without a risk for the surrounding areas.

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Attachment Description of proposed corrective actions

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Abbreviations

Abbreviation	Description
AAC	Alternative Alternate Current
AC	Alternate Current
ATWS	Anticipated Transient Without Scram
CDF	Core Damage Frequency
CNTN	Containment
CR	Control Room
DBE	Design Basis Earthquake
DC	Direct Current
DE	Design Earthquake
DG	Diesel Generator
DID	Defence-In-Depth concept
DSR	Detailed Seismic Zoning
EC	Emergency Commission
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
ECW	Essential Cooling Water
EDMG	Extensive Damage Mitigation Guideline
ELI	Lipno Hydro Station
ELS	Essential Load Sequencer
EOPs	Emergency Operating Procedures
EP	Emergency Plan
EPZ	Emergency Planning Zone
EPZ	Emergency Planning Zone
FAS	Fixtured Alarm System
HA	Hydro Accumulator

Abbreviation	Description
HCLPF	High Confidence on Low Probability Failure
HELB	High Energy Line Break
HP	High Pressure
HPME	High Pressure Melt Ejection
HVAC	Heating, Ventilation and Air Conditioning
HZSp	Fire Rescue Brigade at NPP Site
I&C	Information and Control
IAEA	International Atomic Energy Agency
INPO	Institute for Nuclear Power Operators
IOHO	Internal Emergency Response Organization
IPSART	International PSA Review Team
IZS	Integrated Rescue System
LBB	Leak Before Break
LERF	Large Early Release Frequency
LOCA	Loss Of Coolant Accident
LOOP	Loss of Offsite Power
LOSP	Loss of Safety Power
LP	Low Pressure
MCCI	Molten Core Concrete Interaction
MCR	Main Control Room
MDE	Maximum Design Earthquake
MSK-64	International rating scale for earthquake
N/A	Not Applicable
NPP	Nuclear Power Plant
OHO	Emergency Response Organization
OSART	Operational Safety Review Team

Abbreviation	Description
PACHMS	Post-Accident Containment Hydrogen Monitoring System
PAMS	Post Accident Monitoring System
PGA	Peak Ground Acceleration
POHO	Stand-by Emergency Response Organization
PRPS	Primary Reactor Protection System
PRZR	Pressurizer
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Removal System
RPV	Reactor Pressure Vessel
RTARC	Real Time Accident Release Consequences
SA	Severe Accident
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SEOPs	Shutdown Emergency Operating Procedures
SFP	Spent Fuel Pool
SG	Steam Generator
SMS	Seismic Monitoring System
SONS	State Office for Nuclear Safety
SSAMG	Shutdown Severe Accident Management Guidelines
SSC	Structures, Systems and Components
SV	Safety Valve
TSC	Technical Support Center
UHS	Ultimate Heat Sink

Abbreviation	Description
US NRC	United States Nuclear Regulatory Commission
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association
ZKZ	Modification

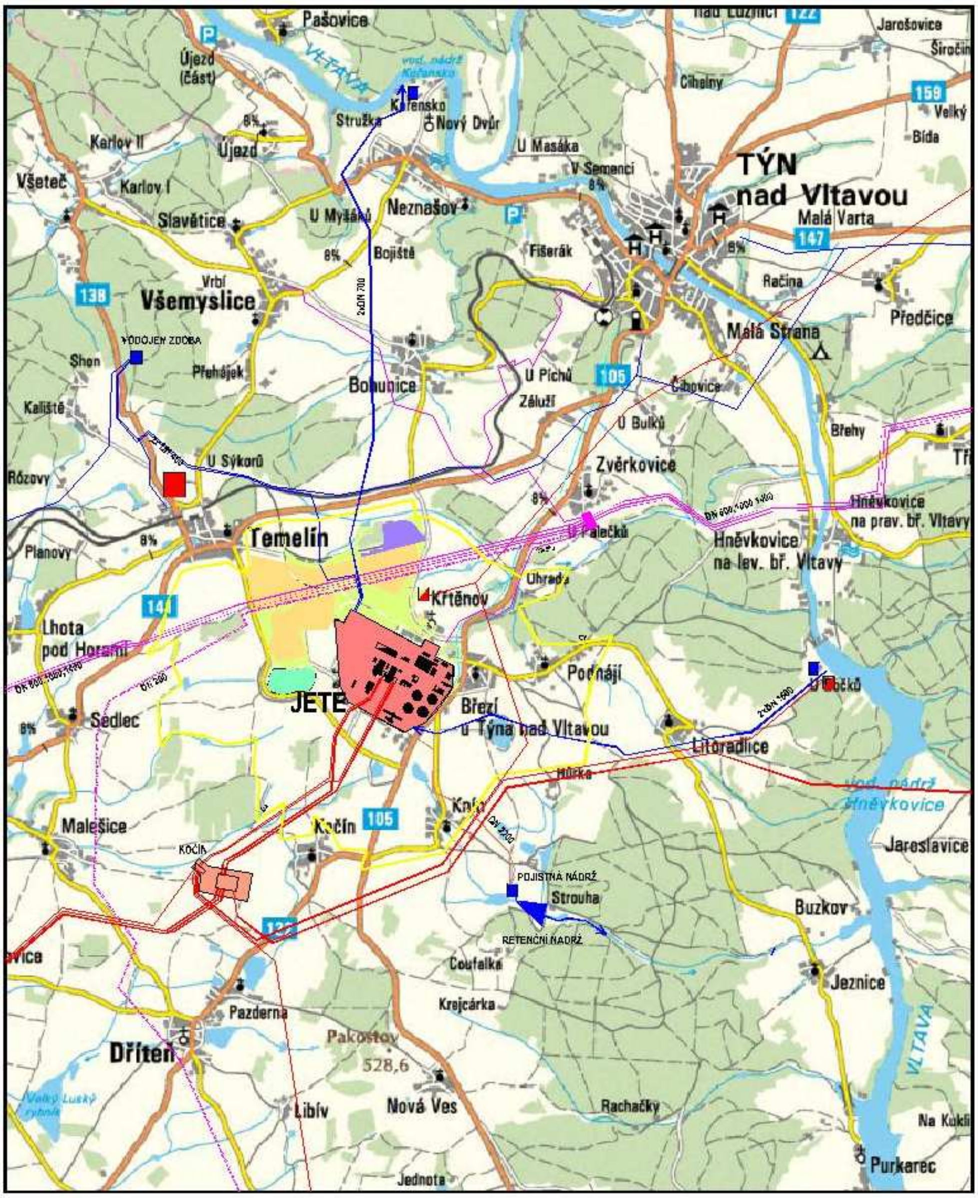
1 General data about the sites and nuclear power plants

1.1 Brief description of the sites characteristics

1.1.1 Site characteristics

Temelín nuclear power plant is situated in South Bohemia, about 25 km north of České Budějovice, at an altitude of 510m above sea level.

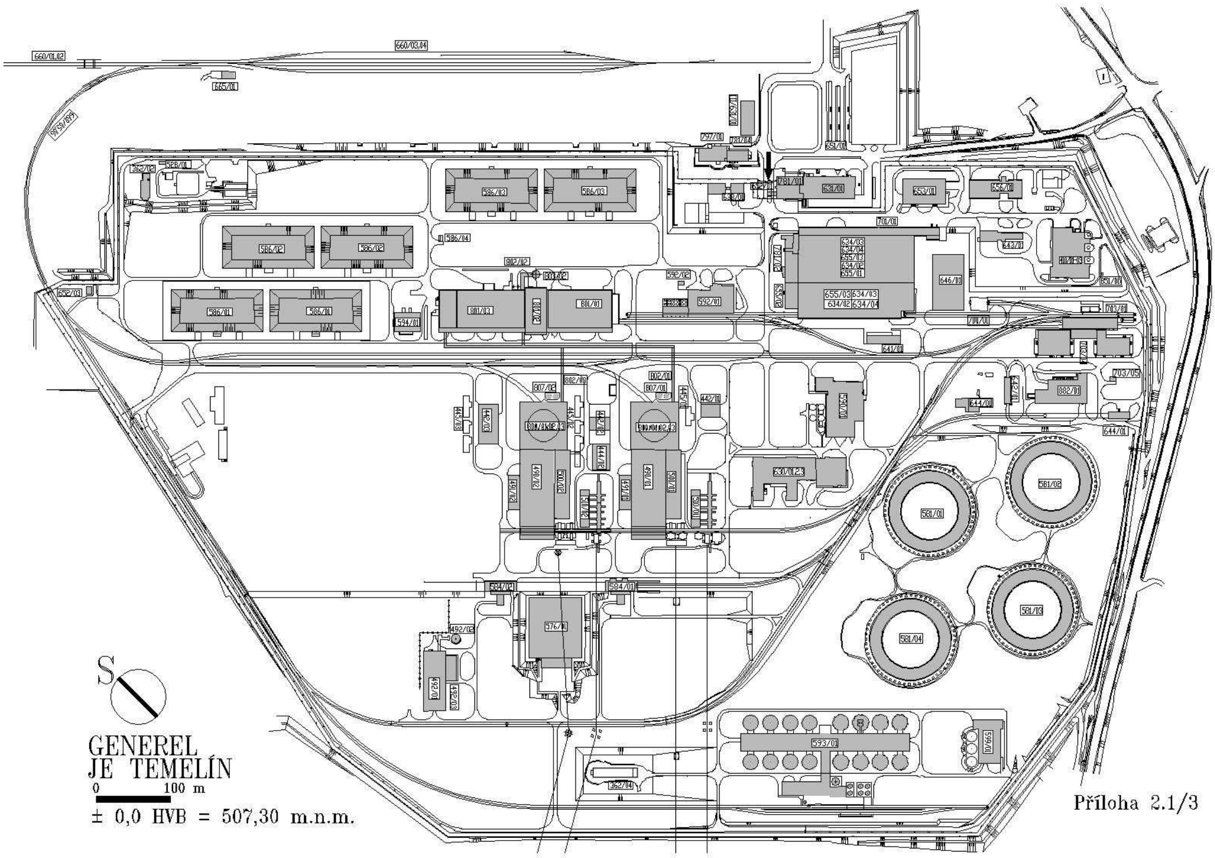
Fig. 1.1.1-1 NPP siting



The nuclear power plant comprises two nuclear units with pressurized water reactors. The closest town is Týn nad Vltavou, located 5km far, to the northeast of the power plant. The power plant takes water for technological purposes from the Hněvkovice Reservoir on the Vltava River (about 5km east of the site). The ultimate heat sink is the atmosphere.

The NPP site also hosts fresh nuclear fuel storage and spent nuclear fuel storage facilities. Spent nuclear fuel is stored in CASTOR type casks, cooled by natural air circulation in spent fuel storage facility. Because of the passive type of cask cooling, there is no threat of loss of the ability to perform the safety function if an initiation event occurs, and spent fuel storage facility therefore is not subject to this evaluation of safety and safety margins.

Fig. 1.1.1-2 NPP layout



The holder of the license for operating all nuclear facilities located on the site is ČEZ a.s., Duhová 2/1444, 140 53 Prague 4. The currently valid operation licenses were issued for unit 1 in Decision of SÚJB No. 22888/2010, issued on 4 October 2010, and for unit 2 in Decision of SÚJB No. 19173/2004, issued on 11 October 2004. Both licenses are valid for 10 years.

1.2 Main characteristics of the units

1.2.1 General unit description

The nuclear power plant comprises two nuclear units with pressurized water reactors VVER-1000 of the series type V 320, each with the nominal power of 3,000MWt. The primary circuit consists of a reactor, a pressurizer and four reactor cooling loops, each with a main coolant pump and a horizontal steam generator.

The primary circuit equipment is placed in a hermetic envelope (containment), made from pre-tensioned concrete. The protective envelope consists of a cylindrical structure with an inner diameter of 45 m, closed by a hemispherical cover. The inner surface of the containment is covered by hermetically tight steel lining. Inside the containment there are also spent fuel storage pools, where spent fuel is transported from the reactor core. To reduce its residual output, the spent fuel is put in a cask and taken to the spent nuclear fuel storage (its capacity corresponds with the power plant's lifetime).

The reactor (specifically the reactor core) is cooled and moderated by light primary circuit water, pumped through the core by means of main coolant pumps. After passing through the reactor, the heat accumulated in the coolant is transferred to secondary circuit water in steam generators. The pressure in the primary circuit is maintained by a pressurizer.

The secondary circuit comprises steam generation equipment (secondary SG side), a feed water system, one turbine generator with the nominal electrical power of 1,000MWe and a regeneration system.

The active safety systems are provided with the 3 x 100 % redundancy and are mutually independent and physically separated. The passive safety systems (the hydro accumulators inside the containment) are provided with 2 x 100 % redundancy. Seismic resistance of all the redundant safety systems is provided, including the power supply and the control systems as well as all the other auxiliary systems. The emergency power supply sources of the power supply systems and the control systems are mutually independent, physically separated and seismic resistant (subject to qualification applicable to safety systems). In addition there are back-up non-seismic resistant power supply sources for the safety-related systems. The design provides diversification of systems for the fulfillment of three fundamental safety functions - 1) provision of the reactor shutdown (reactivity control), 2) heat removal from nuclear fuel (cooling down) and 3) prevention of radioactive releases (barriers and the containment isolation) - see chapter 1.3.

Fig. 1.2.1-1 Main SSC

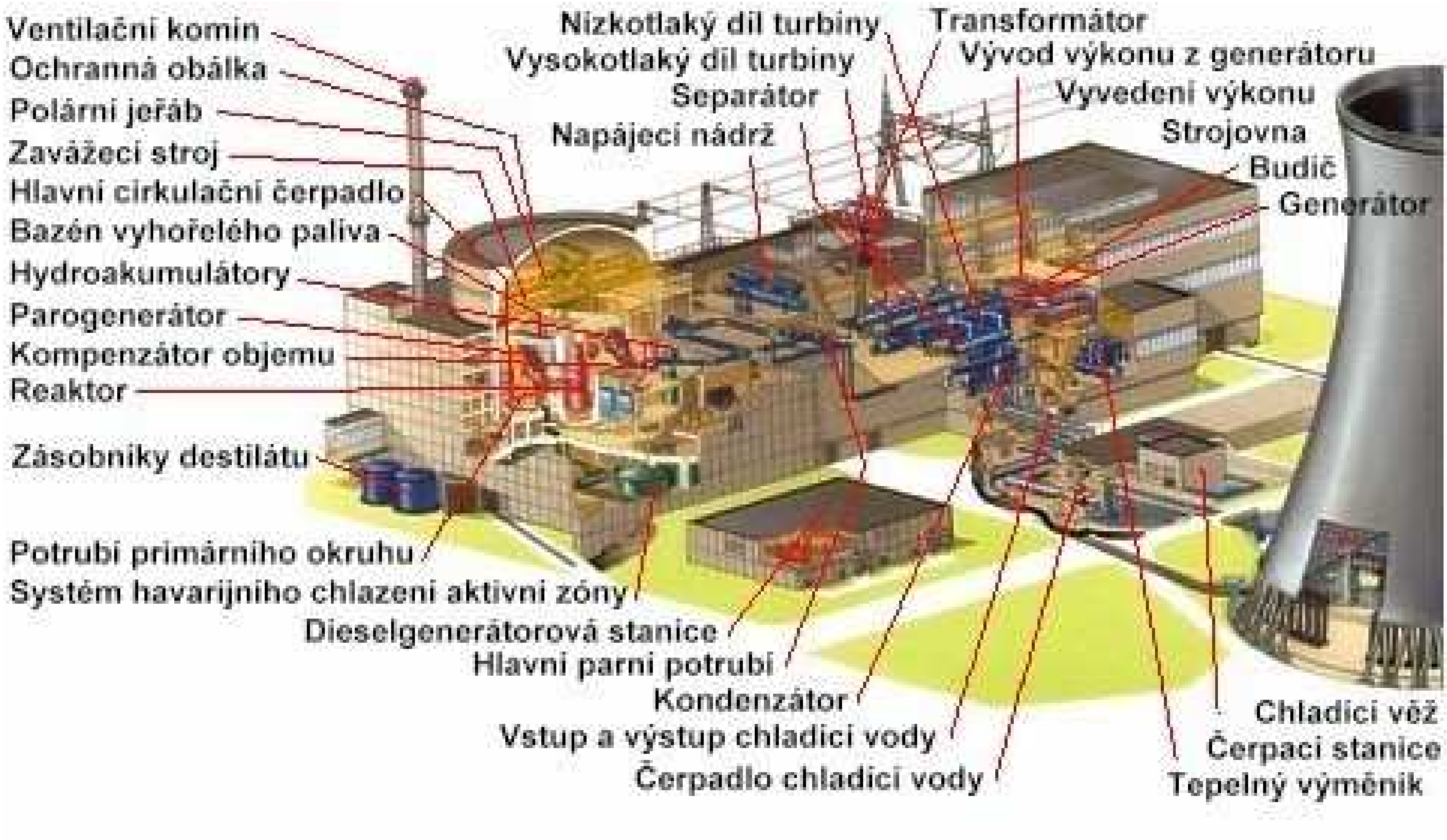
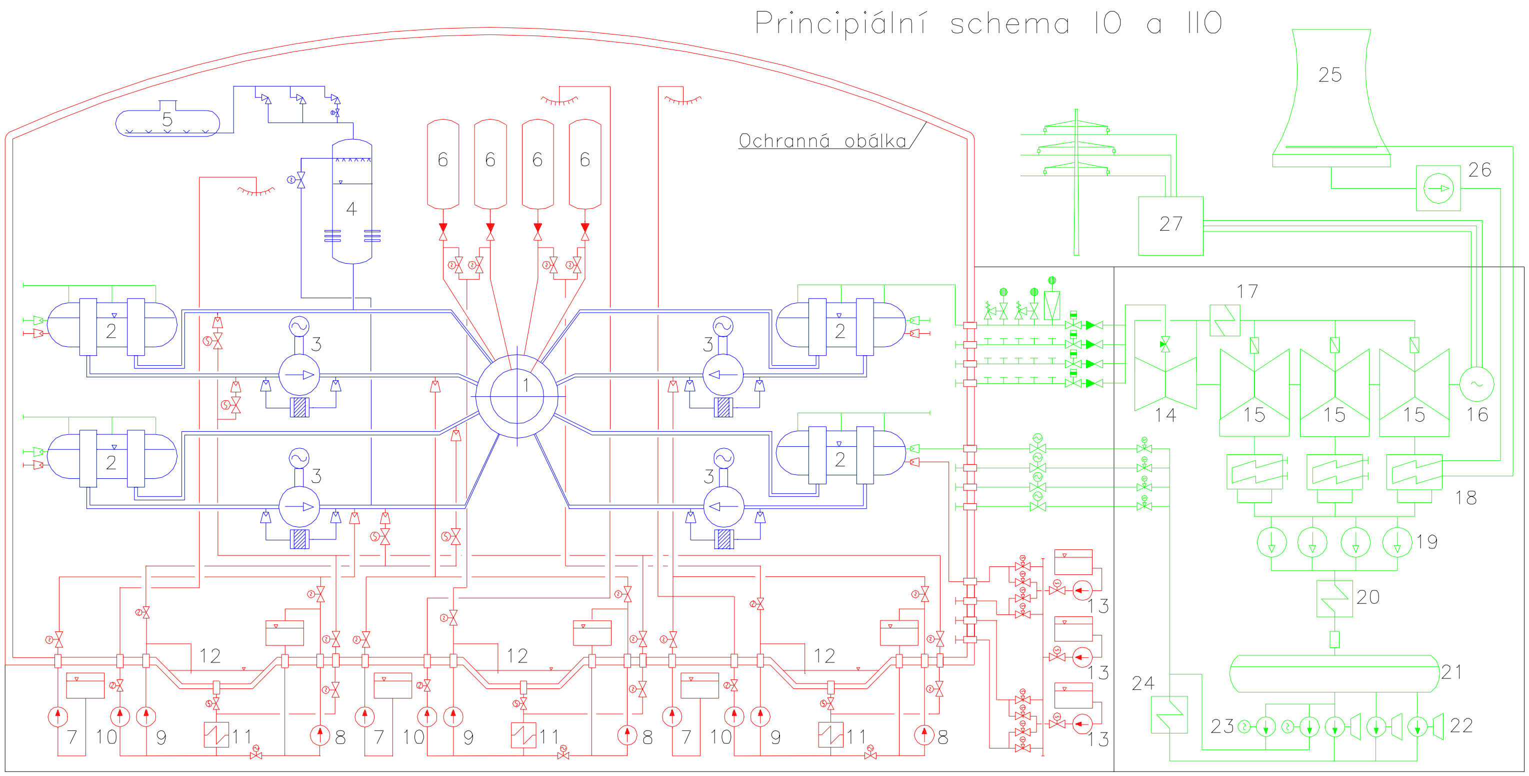


Fig. 1.2.1-2 Main NPP system diagram



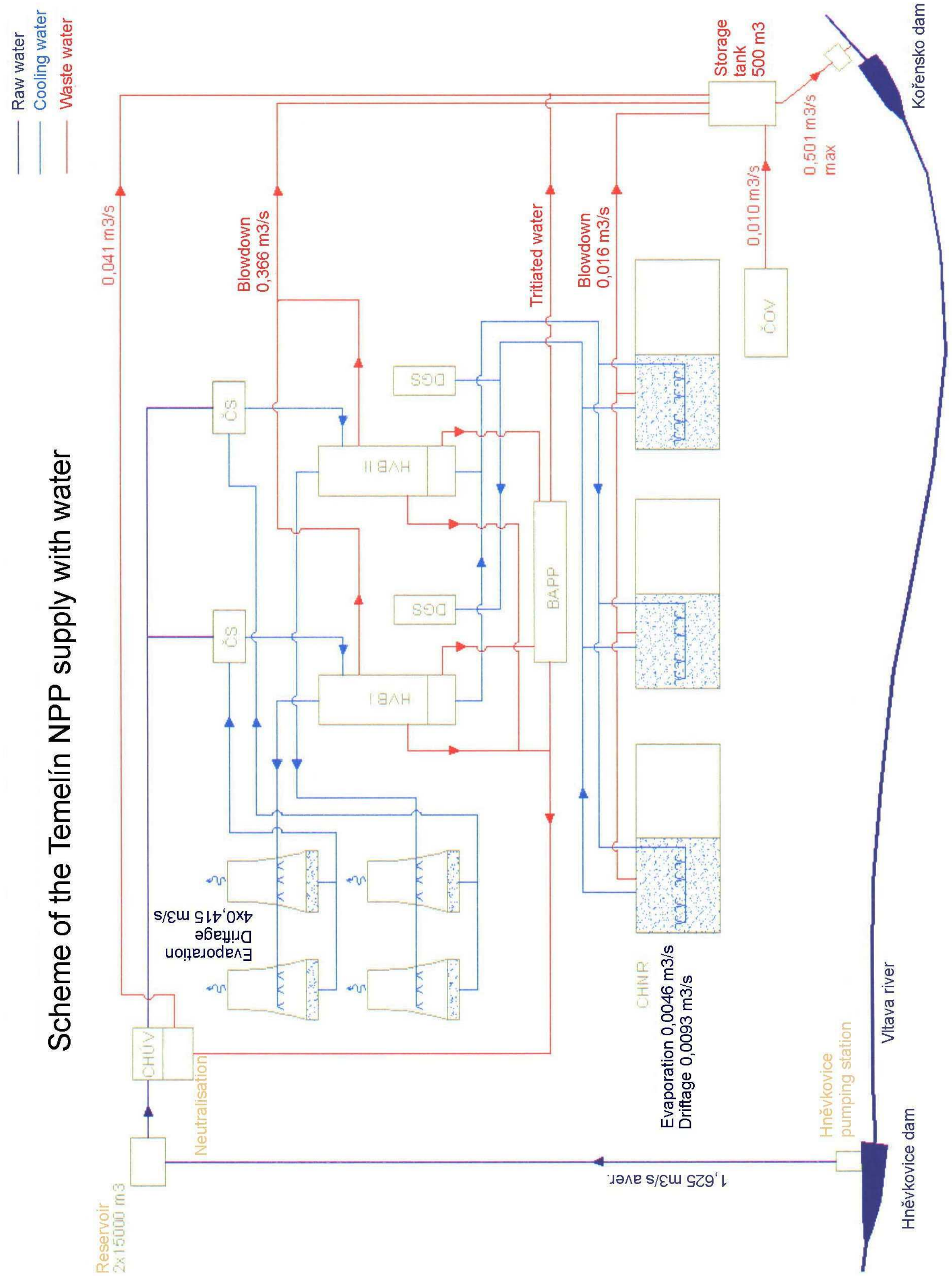
1	Tlakovodní reaktor
2	Parogenerátory
3	Hlavní cirkulační čerpadla
4	Kompenzátor objemu
5	Barbotážní nádrž
6	Tlakové zásobníky
7	VT doplňovací čerpadlo
8	Havarijní vstřikovací čerpadlo
9	Havarijní dochlazovací čerpadlo

10	Sprchové čerpadlo
11	Tepelný výměník HSCHZ
12	Nádrž zásoby kyseliny borité (GA201)
13	Havarijní napájecí čerpadlo
14	Vysokotlaký díl turbíny
15	Nízkotlaké díly turbíny
16	Generátor
17	Separátor – přihřívák
18	Kondenzátor

19	Kondenzátní čerpadla
20	NT regenerace
21	Napájecí nádrž
22	Turbonapaječky
23	Elektronapaječky
24	VT regenerace
25	Chladící věže
26	Čerpací stanice
27	Transformátor vyvedení výkonu

Obr. 3

Fig. 1.2.1-3 NPP water supply



The power plant takes water for technological purposes from the Hněvkovice Reservoir on the Vltava River (about 5km east of the site). The ultimate heat sink is the atmosphere. In normal operation, decay heat is removed to the atmosphere through cooling towers (two per unit); in accident conditions, this is done through steam generators and a steam dump to the atmosphere or through the essential service water system and the coolant tank with spraying.

The site is connected to the off-site power grid by means of two 400kV lines and two 110kV lines through the Kočín substation.

For the first time, controlled fission reaction (criticality) was achieved on unit 1 on 11 October 2000 and on unit 2 on 31 May 2002.

1.2.2 Main Safety Modifications

Based on results of independent international missions, expert opinions, suggestions of Czech experts and SÚJB requirements, number of technical modifications of the original design have been proposed, which guarantee fulfillment of western NPP standards. Among the important safety modifications belong:

- Replacement of the instrumentation and control system, including new design (Westinghouse).
- Replacement of the nuclear fuel, including the core design (Westinghouse).
- Replacement of the original radiation monitoring system, including the design.
- Replacement and complementation of the diagnostic system.
- Replacement of the original cables to non flammable and non fire proliferation cables.
- Significant modifications of the electric systems (electric protections, additional common dieselgenerators, increased capacity of batteries, etc.).
- Protection of high energy piping at elevation of +28,8 m – installation of whip restrictors, application of the „no break zone / superpipe“ concept.
- Qualification of components important for safety, in particular the safety valves (steam dump to atmosphere functional qualification, SG SV qualification for water flow and steam-air mixture flow, PRZR SV and PRZR relief valves qualification for work with water media).
- Pressure vessel integrity and pressure-temperature shock, program of long term evaluation of surveillance samples, pressure-temperature shock analysis.
- Primary loop components integrity - non destructive inspection qualification.
- Installation of local seismological network.

- Implementation of the beyond design basis accidents management.
- Redesign of core, change of nuclear fuel supplier (TVEL).
- Primary circuit protection against overpressure at cold state.

Periodic Safety Review:

In the period between 9/2008 – 9/2009 PSR took place in Temelín NPP. The process of Temelín NPP PSR has been prescribed to maintain the general compliance with IAEA NS-G-2.10 instruction and WENRA principles for PSRs, defined in "Reactor Safety Reference Levels". NPP PSR has been implemented mostly by ČEZ, a.s., which is in compliance with IAEA NS-G-2-10 instruction.

In accordance with IAEA NS-G-2.10 the objectives of PSR, by thorough review of the status of key areas affecting safety, were:

- To identify in what scope the power plant complies with current internationally recognized safety standards and practice.
- To verify validity of licensing documentation.
- To identify if particular measures for keeping the power plant safe until next PSR are applied.
- To identify areas for improvement in the safety area, that should be implemented to solve identified safety deviations.

For PSR purposes documentation has been prepared in advance, including in particular Methodologies and Criteria, resulting from Czech legislative documents and IAEA documents up to the level of Safety Guide, INSAG series documents and WENRA requirements for PSR. All (14 areas) and all safety factors defined in compliance with IAEA NS-G-2.10 instruction were subject to the review.

Comprehensive review within PSR identified similar areas for safety improvement as those included in this report. Some of them are currently in the phase of preliminary design preparation and would be implemented regardless this new review. They include in particular improvement of NPP design resistance against consequences of severe accidents (including improvement of hydrogen disposal system capacity, localization of core melt in ex-vessels phase (outside of the reactor pressure vessel) and alternative water feeding to the containment sump. In administrative and personal areas it is creation of SAMG and PSA Level 2 for shut down statuses and simulator modifications resulting in possibility to simulate conditions of transition from emergency conditions to the first phases of a severe accident as a personnel training tool. Implementation of the proposed measures in these areas is expected in NPP at latest until end of 2018 (until next PSR).

1.3 Systems for providing or supporting main safety function

Safety of the NPP is provided by its ability to meet the following general safety criteria:

1. Shutdown the reactor in a safe manner and to maintain it under the conditions of safe shutdown.
2. Remove residual heat from the reactor core and from spent fuel.
3. Confine releases of radioactive media so that the release does not exceed the set limits.

Meeting the aforementioned general safety criteria is based on application of the following principles

- Defense in depth
- Fulfillment of safety functions

Defense in depth

Defense in depth has two principal assignments:

- Prevention of accidents
- Mitigating consequences of incidents (accidents)

The fulfillment of defense in depth within the JE project is realized by means of the following provisions:

- Five levels of defense in depth
- Three protection barriers to prevent release of radioactive media to the environment (fuel and its cladding, pressure limit of the primary circuit, containment)

Goals of the five levels of defense in depth:

1. Prevent deviations from normal operation and failures of the systems.
2. Find and remove deviations from normal operation to prevent transformation of the assumed operating conditions into accident conditions.
3. In case of occurrence of accident conditions, to prevent development of failures and incidents into beyond design basis accidents and to retain radioactive media within the containment using the safety systems.

4. Maintain integrity of the containment.
5. Mitigate radiological consequences of radioactive media release to the environment.

In case of failure of one level of the defense in depth, the following level thereof is activated.

Performance of safety functions

Functions, namely both the operational and the protection and safety ones, are fulfilled by the technological systems, structures and components. All SSC are classified in accordance with IAEA standards as safety systems, safety related systems and systems not-important for safety.

Form the point of view of nuclear safety, SSCs of the Dukovany JE are divided as follows:

- Important from the point of view of nuclear safety (participating in the fulfillment of safety functions).
- Unimportant from the point of view of nuclear safety (do not fulfill any safety function).

The systems important from the point of view of nuclear safety are further divided, based on their function and importance for nuclear safety, as follows:

- Safety systems
- Safety-related systems.

Safety systems represent a set of systems which include the following:

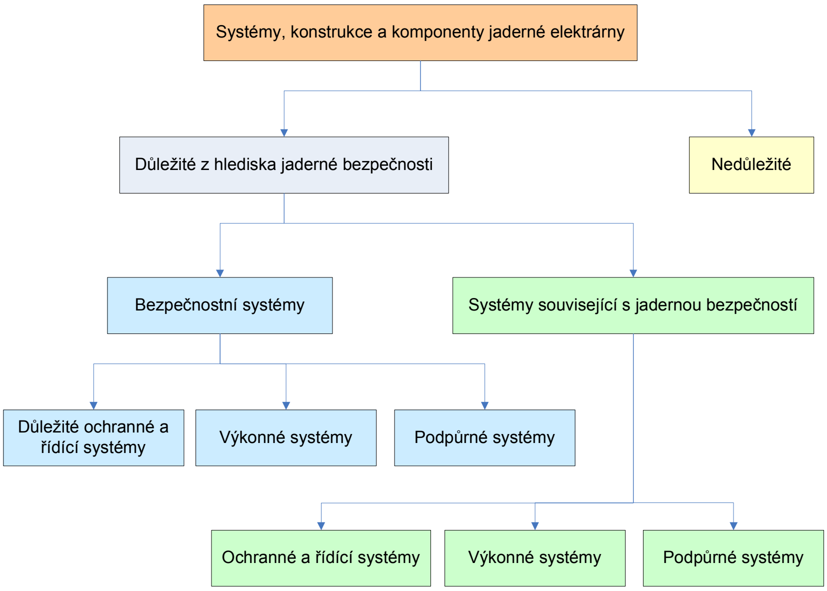
- Important protective and important control systems (instrumentation for measurement, or monitoring of quantities or conditions important from the point of safety and automatic activation of the respective systems with a view to providing and maintaining the unit in a safe condition).
- Performance safety systems (systems which perform the respective safety functions by initiation of the safety systems).
- Auxiliary systems (systems providing for the functions of the protective and performance systems, such as power supply, cooling systems, etc.).

Nuclear safety-related systems represent a set of systems which include the following:

- Protective and control systems.
- Performance systems and structures.
- Auxiliary systems (power supply, cooling systems, etc.).

Unimportant systems do not fulfill any safety function, however, may be used in case of accident conditions (if they are available).

Fig. 1.3-1 SSC classification into individual categories



The systems important from the point of view of nuclear safety, i.e. the safety systems and the safety-related systems, are included in the classified equipment and are, in compliance with the applicable legislation requirements, divided into three safety classes depending on their importance from the point of view of safety.

The technological systems, structures and components are classified from the point of view of seismic resistance as well. All the safety systems (and a part of the safety-related systems) are classified in the first category of seismic resistance.

The safety systems are subject to the qualification process.

Active safety systems are arranged in 3 divisions of safety systems backed up a whole (the concept of 3x100% redundancy). In accordance with this concept there is a system of essential power supply in each division, serving as a supporting safety system for power supply for appliances of the given division. To secure the necessary degree of redundancy, these safety systems are independent and separated from each other as far as their layout (structurally as well as from the point of view of fire protection), their electrical arrangement and the control system are concerned. Each system includes its own emergency sources (DG, accumulator) and power distribution lines. Considering the principle of independence and separation a single failure of one system can not affect availability of the remaining two divisions.

The passive safety systems are based on the assumption of possible functional failure of some of the passive system component and therefore redundancy of this system follows the

100+100% principle. They consist of two independent, functionally identical and separated subsystems.

The ability to ensure basic safety functions by means of the following diversified multiple systems in normal and abnormal operation and in accident conditions is typical for units of this type:

- Subcriticality during at power operation (under normal and abnormal operating conditions) is ensured by passive and active means - mechanical control (dead-weight fall of clusters to the core) and by normal boron make-up system (safety-related systems). Under emergency operating conditions if normal operating means are not available, safety systems are used - high-pressure ECCS with a high concentration of boric acid, low-pressure ECCS and passive safety systems with a shut-down concentration of boric acid.
- Heat removal at power operation (under normal and abnormal operating conditions) is ensured by horizontal steam generators with a large water inventory; that produce steam for turbine. Heat removal at low power is ensured by steam dumps to the condenser, a reduction station and auxiliary condenser - all classified as safety-related systems. Heat removal from the turbine condensers to UHS is provided by the coolant circuit with cooling towers with natural draught. Heat removal for emergency operating conditions is ensured by safety systems - steam dump to the atmosphere and, if applicable, relief valves of steam generators, an low-pressure ECCS with heat removal by means of ECW, with a redundancy of 3 x 100 % and an internal redundancy of 100 + 100% for active components (pumps) inside each of three equal divisions. Heat removal from ECW to the atmosphere is ensured using coolant tanks with spraying, also with the 3 x 100% concept.
- Release of fission products from the core is prevented by physical barriers – matrices and fuel cladding, the pressure boundary of the primary circuit, the full pressure containment (app. 60 000 m^3) with underpressure maintained in it under normal and abnormal conditions. Under normal and abnormal conditions, heat removal from the spent fuel pool is provided by a redundant (3 x 100%) safety-related system, and under accident conditions by a safety system with coolant supply from the safety containment spraying system, with evaporation to the containment in case the coolant circulation system is unavailable. Under accident conditions (LOCA or HELB), containment isolation from the surroundings is also actuated by closing quick-acting valves on the boundary of the containment with heat removal and by depressurizing

the containment using an active safety containment spraying system (redundancy 3x100 %) with heat removal using ECW to the coolant tanks with spraying.

Detailed descriptions, including the method of solution of beyond design-basis conditions, are specified in the following chapters.

Compliance of the plant with the current licensing basis

Legislative requirements for NPP design in area of nuclear safety, radiation protection and emergency preparedness are included in SONS Act No. 195/1999. NPP design meets the requirements of the aforesaid Act and the analyses proved that during all expected normal, abnormal and emergency operating conditions the limit nuclear fuel parameters are not exceeded and during design bases accidents the exposure of critical group of inhabitants will not reach the conditions included in SONS Act No. 215/1997 and the early safety measures initiation and completion would not be feasible.

Fuel damage is not admissible for all types of design accidents. This is proved by fulfillment of safety criteria following various mechanisms of fuel rods damage and determining limit values of criteria parameters corresponding to these mechanisms (parameters according to which fulfillment of acceptability criteria is verified), when it is guaranteed that fuel rods will not be damaged in case the limit values will be exceeded.

In order to prove safety criteria the initiating events are classified within 4 categories:

Category I - normal operation with admissible deviations and operation transients

The Category I events include those which are expected to occur often or on regular basis during power operation, refueling, maintenance and during planned power changes. Category I events as such are managed with a well margin between any plant parameter and the value of such parameter which would trigger a protective action.

Category II - accidents (events) of low frequency of occurrence (normal operation interruption)

In the worse case such accidents lead to reactor trip and under all circumstances the power plant is capable to return to normal operation after the Category II event.

Category III - accidents (events) of rare occurrence (design basis accidents)

According to the definition the Category III accidents include those which may occur very rarely during the life of the power plant.

According to the definition the Category III events include those which may occur rarely during the life of the power plant, with the probability of occurrence not exceeding 10^{-2} /reactor-year.

Category IV - limiting (design basis) accidents (events)

Category IV accidents are such accidents, which are not expected to occur, however they are discussed since their consequences would include potential releases of significant amount of radioactive material. These are the most serious accidents, for which the plant must be designed, and represent the limiting design cases.

It was proved by safety analysis that for **Category II** accidents not exceeding of the limits set for safety operation of fuel rods is assured from the point of view of radiation safety, not exceeding of the safe operation limit with regards to the number and size of fuel rod defects.

For **Category III ÷ IV** accidents fulfillment of fuel system safety requirements has been proved, and according to that:

- core cooling as well as post-accident core unloading are assured,
- the fuel system is not damaged in a scope preventing normal operation of control rods,
- heat removal from the fuel system to the ultimate heat sink is always provided,
- number of damaged fuel rods is evaluated conservatively for the purpose of radiological consequences.

1.3.1 Reactivity control

1.3.1.1 Core subcriticality

The reactor core is designed in away to ensure that the resulting effect of immediate feedbacks in the core acts against quick increase of reactivity under all operation modes with critical reactor. The reactor and the core are designed to operate with negative value of reactivity temperature coefficient. Therefore increase of moderator temperature leads to reduction of system reactivity, resulting in tendency of the moderator temperature to return to its original value. Reactivity increase, which results also in increase moderator temperature, is therefore power limited and leads to establishment of stable operating conditions.

Reactivity control operating systems

Two independent systems, based on different technical principles, are provided for reactivity control:

- Mechanical system of reactor control and shut down including control rods drive mechanisms breakers.

- Normal boron acid make-up and control system (chemical and volume control system).

The mechanical reactor control and shutdown system is classified as a safety system.

Reactor protection provides for fall of all control rods into the core in order to stop the chain reaction.

Fall of all control rods is provided by power supply switch off. Control rod means a bunch of 18 rods, absorbing neutrons.

Control rods have the following functions:

- They provide for reactor trip - interruption of the chain reaction in the reactor by their fall into the core.
- They contribute to automatic control with the objective to maintain reactor power at specified level and assure transition from one power level to another.
- They compensate for quick changes in reactivity (power and temperature effect).

There are 61 control rods in total, split into 10 groups. The control rods are grouped into 6 groups intended for reactor shut down (they are in fully withdrawn from the core during operation) and 4 control groups (they move up and down with an overlap and can be controlled both manually and automatically). All groups contribute to the quick reactor shut down functionality by their fall to the core by gravitation. The system consists of the following main components:

- Breakers of (AC and DC) power supply as part of the switchboard.
- Control rods

The control rods are installed in the reactor head, situated in the reactor building inside the containment, the breakers are installed in the electric switching station in the reactor building.

The system of normal RCS make up and letdown and boron control is classified as a safety related system.

This system serves for RCS make up and control of boron acid concentration in RCS coolant. To reduce boron acid concentration in RCS part of the coolant is letdown through filters and make up deaerator to the impure condensate tank. Pure condensate through the boron control deaerator is used for RCS make up. RCS boric acid concentration is increased by injection of boric acid to make up pump suction. Make up water heating in the deaerators is provided by heating steam from the turbine hall. In addition the system provides for cooling and letdown of coolant from the primary circuit to treatment and return of the treated coolant back to the RCS through make up deaerator. The system also provides for RCP seal injection.

The RCS make up and letdown system form one technological circuit. The make up pumps are designed as redundant based on the 3x100 % principle, while two of them are supplied from the essential power supply system (each from different common DG).

The components and piping are located outside and inside the containment. The regenerative heat exchanger of the primary circuit coolant letdown is situated inside the containment. The make up pumps, the make up deaerator, the boron control deaerator and the deaerator steam cooler are situated outside of the containment. Also the make up water cooler, the after-cooler, the pump coolers and the condensate cooler are situated outside of the containment.

The make up pumps, consisting of a couple of serial pumps, are situated in separate rooms, i.e. they are physically separated from each other.

For operation of the system of normal make-up an oil system is required as a supporting system. Each make up and boron control pump has an autonomous oil system, including pumps, coolers, filters and tanks. Oil is pumped to the cooler from the oil tank by a pump. Part of the oil from the cooler flows directly to the hydraulic clutch of the pump and part flows through the filter to the bearings and the gear and from there back to the tank. Make up pump oil system components are situated in the reactor building outside of the containment, provided that for each pump the oil system is installed in independent rooms, which are separated and therefore the oil systems are independent of each other. Power supply of oil pumps is designed analogically to the power supply of the make up pumps.

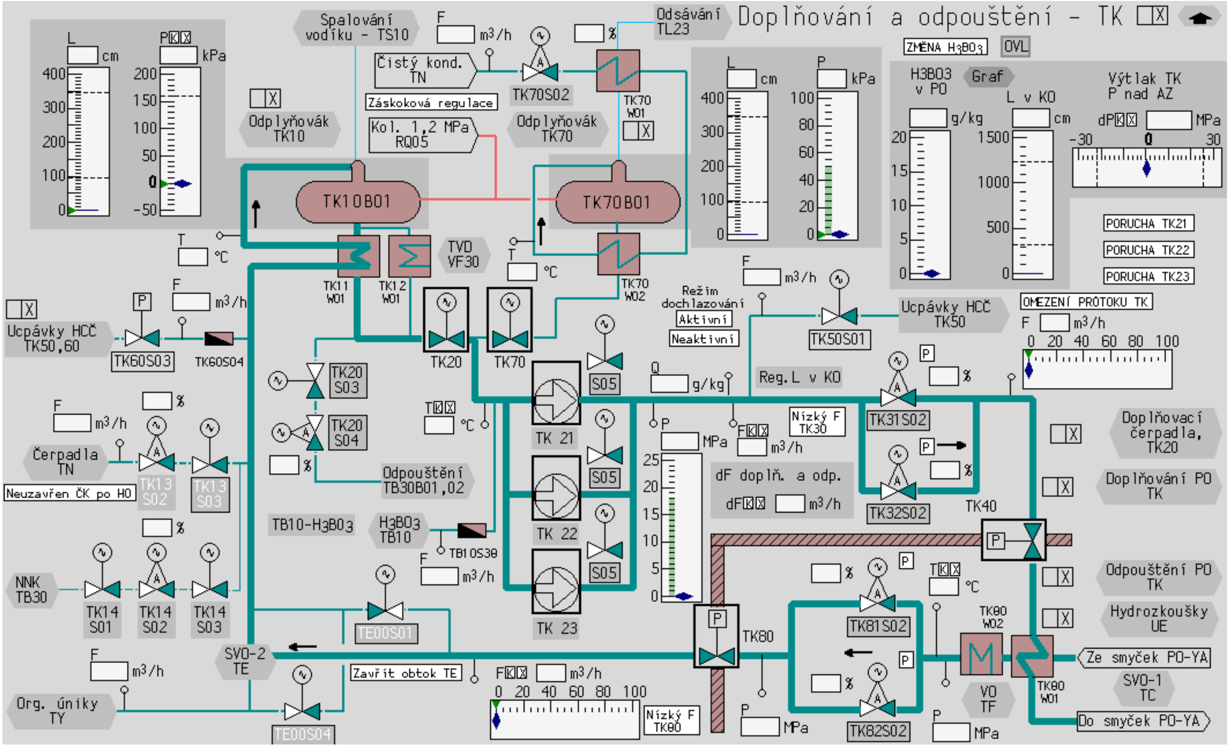
Additional auxiliary and supporting systems serve for boric acid supply for primary circuit make up, in particular the boric acid system, which is also classified as a safety related system.

The purpose of this system is storage of H_3BO_3 concentrate and its delivering to the RCS make up system. The system consists of two storage tanks for H_3BO_3 concentrate and three pumps. The H_3BO_3 concentrate is brought to each of the two storage tanks through separate routes, one from the concentrate treating station and one from the chemical reagent preparation plant. From the tank the concentrate (40g of H_3BO_3/kg) is delivered by pumps to RCS make up pump suction.

Alternatively boric acid solution may be delivered to the RCS make up pump suction from impure condensate tank through normal make up deaerator.

The boron concentrate tanks and pumps are situated in the rooms outside of the containment.

Fig. 1.3.1-1 Normal RCS make up system



Safety systems

In addition for reactivity control safety function there are the high pressure emergency core cooling system and the emergency boron injection system available, both classified as safety systems.

These systems are designed for mitigation of the course and consequences of accidents associated with RCS leakage, or possible secondary circuit leakage (LOCA or HELB). During normal operation of the unit at nominal or reduced power these systems are in standby mode, ready to act in case of emergency. In emergency, as far as reactivity control is concerned, these systems prevent unacceptable transient processes associated with reactivity changes.

The systems are designed with 3x100 % system redundancy, including all supporting systems (cooling, power supply, control and ventilation).

The high pressure boron injection system consists of three piston pumps and the high pressure emergency core cooling system consists of three centrifugal pumps. Each pump has its own tank with a inventory of concentrated boric acid solution (40g/kg). All pumps are supplied from the essential power supply system (DG).

The system is located in the reactor building outside of the containment and the delivery pipes pass through to the containment boundary and are connected to the primary circuit. The pumps are installed in independent, physically separated rooms of the safety systems.

The high pressure ECCS tanks are situated inside the containment in separated compartments, high pressure emergency boron injection tanks are situated outside of the containment in separated rooms.

In addition, in order to provide for core subcriticality, other inventory of boric acid is available in the containment sump, common for HP (high pressure), LP (low pressure) ECCS and the containment spray system. Sufficiently high boric acid concentration is present also in the hydro-accumulators, which form passive system of emergency core cooling.

1.3.1.2 Spent fuel pool subcriticality

Subcriticality of spent fuel assemblies stored in the spent fuel pool is provided using two independent methods:

- The geometry and material composition of the storage grid installed in the spent fuel pool.
- By boric acid concentration in the pool coolant.

Subcriticality is provided even in case the spent fuel storage pool is filled with pure condensate.

1.3.2 Heat transfer from reactor to the ultimate heat sink

For VVER1000 units at Temelín 6 operation modes are classified.

Tab. 1.3.2-1 Operation modes

Operating mode	Title	Heat output	Mean RCS temperature	k_{ef}
1	Power operation	$\geq$2 % N_{NOM}	> 260 °C	$\geq$ 0,99
2	Zero power operation	< 2 % N_{NOM}	> 260 °C	$\geq$ 0,99
3	Hot state	Decay	> 260 °C	< 0,99
4	Semi-hot state	Decay	260 °C >$T_{stř}$ $\geq$150 °C	< 0,99
5	Cold state	Decay	150 °C > $T_{stř}$ $\geq$70 °C	< 0,99
6	Outage	Decay	< 70 °C	< 0,98

At certain plant configuration with shut down reactor the probability of core damage may be higher than during unit operation at full power. From all unit operation modes the most risky configuration is loss of residual heat removal system during mode 6 with the reactor level in the middle of cold legs nozzles (mid-loop operation).

During mid-loop operation, with nuclear fuel present in the core, special attention must be paid to core cooling. Reactor level decrease is permitted only after reduction of residual power to a level, which can be safely removed by the residual heat removal system. In this operating mode (soon after reactor shut down in particular) there is a high residual power in the primary circuit, in combination with small volume of coolant, and in addition to that there are certain organizational limitations, such as repairs or maintenance of some components, or non existence of the control system automatic action. Therefore, in case of loss of cooling or uncontrolled loss of coolant, there are minimum time margins set for cooling recovery, or possibly for leakage detection and elimination. Therefore with reduced level in the reactor number of technical and administrative limitations exist in order to reduce the risk of loss of residual heat removal to minimum (provision of sufficient inventory of coolant and technical means for primary circuit refilling, pre-job-briefings prior primary circuit drainage, permanent attention of selected personnel in the control room etc.).

The reactor level decrease for middle-loop operation with nuclear fuel present in the core is performed no earlier than 7 days after reactor shutdown and takes maximum 2 days. In comparison with operation in other operating modes this time is very short.

The following chapters describe method of residual heat removal from the core to UHS in modes 3 to 6, i.e. modes with shut down reactor including open reactor with low water level.

1.3.2.1 Existing heat transfer means

Operating systems (classified as non essential, or safety related systems), as well as other systems, classified as safety systems may be used for heat removal from the core.

Operating systems of heat removal from the core

The normal operating system designed for heat removal from the core to the ultimate heat sink is heat removal through the secondary circuit, provided that heat removal from the core is provided by forced circulation (if the reactor circulating pumps are in operation) or by natural circulation. In the residual heat removal mode steam from SG flows to the turbine condenser and heat from the condenser is removed by the circulating cooling water to the natural draught cooling towers. From the condenser the condensate returns to the feeding tank by the action of the main or auxiliary condenser pumps, from where water is fed to the steam generator by auxiliary or turbine driven feedwater pumps. The auxiliary pumps serve for condensate or feed water make up in modes 2 and 3.

For residual heat removal while maintaining the reactor in hot state the auxiliary condenser could be used. Heat from the auxiliary condenser is removed by the non-essential cooling water to the natural draught cooling towers. During RCS cooldown, after the parameters are decreased below values, when heat removal to the turbine condenser is ineffective, heat removal is provided by LP core cooling system (safety system) which works as a normal operating system in this mode. Heat removal from the RCS is provided in ECCS heat exchangers, from where heat is removed by ECW. ECW transfers the heat to the atmosphere in the spray cooling tanks. Return of the cooled RCS coolant from ECCS heat exchanger back to the primary circuit is provided by LP ECCS pumps.

Safety systems

In case of unavailability of the aforementioned normal operation systems there are other systems available, which are classified as safety systems.

Heat removal from the core is provided by natural or forced coolant circulation. In this case heat removal in the secondary circuit is provided through open circuit, i.e. via the steam dump to atmosphere, or through SG SV to the atmosphere. This mode is called the secondary feed and bleed. In case of unavailability of the auxiliary feed water pump feeding of water to SG is provided by emergency feedwater pumps (3x100 % redundancy).

If heat removal from the RCS can not be applied through the secondary heat removal by discharge to the atmosphere (via the steam dump to the atmosphere, or via SG SV) or in case of coolant leakage from the RCS, there is a possibility of an alternative heat removal from the core using the primary "feed and bleed" method, controlled by coolant injection to the RCS using ECCS and coolant letdown from the RCS to the containment. In this mode heat would be removed through ECCS heat exchanges cooled by ECW and via ECW COOLING POOL to the atmosphere.

Other systems

Use of other systems (beyond the design specification) is described in the chapter 6 Severe Accident Management.

1.3.2.2 Lay out information on the heat transfer chains

Operating systems

Auxiliary feedwater system is used for steam generator feeding during unit start up and shut down. It consists of two electric driven feedwater pumps, with suction from the feedwater tank. Discharge of the pumps is to the common feedwater header of SGs downstream of HP (high pressure) feeding water heaters. Pumps and valves are supplied with power from the essential power supply system (each pump from different common DG). The water source for the auxiliary feedwater pumps is the feedwater tank situated in the intermediate machine hall at elevation of +30.0 m.

For residual heat removal while maintaining the reactor in hot state with power less than 3 % the auxiliary condenser could be used. Steam from the main steam header to the auxiliary condenser is introduced by control valves, heat from the auxiliary condenser is removed by the non-essential cooling water to the natural draught cooling towers and condensate is pumped back to the feedwater tank. The auxiliary condenser is situated in the machine hall at elevation of ±0.00 m.

Residual heat removal system is used during RCS cooldown if RCS temperature is below 150℃ and RCS pressure below 1.7 MPa. If RCS pressu re is greater than 1.7 MPa it is not possible to connect the RHR system due to limit parameters of ECCS heat exchanger. The normal RCS cooldown rate is 30 ℃/hour, the maximum cooldown rate in emergency situation is 60 ℃/hour. The residual heat removal system may be operated in the direct coolant circulation mode, when cold coolant is injected to the cold leg of the circulation loop / to the downcomer and hot coolant is removed from the hot leg of the loop No. 4 to ECCS heat exchanger and to the LP ECCS pump suction. At reduced level in the reactor the residual heat removal system is operated in the reverse coolant circulation mode, when cold coolant is injected to the hot leg of the circulation loop/upper plenum and hot coolant is removed from the cold leg of the loop No. 4 to ECCS heat exchanger (from where heat is removed to the ultimate heat sink by ECW system) and to LP ECCS pump suction.

Fig. 1.3.2-2 SG feedwater system

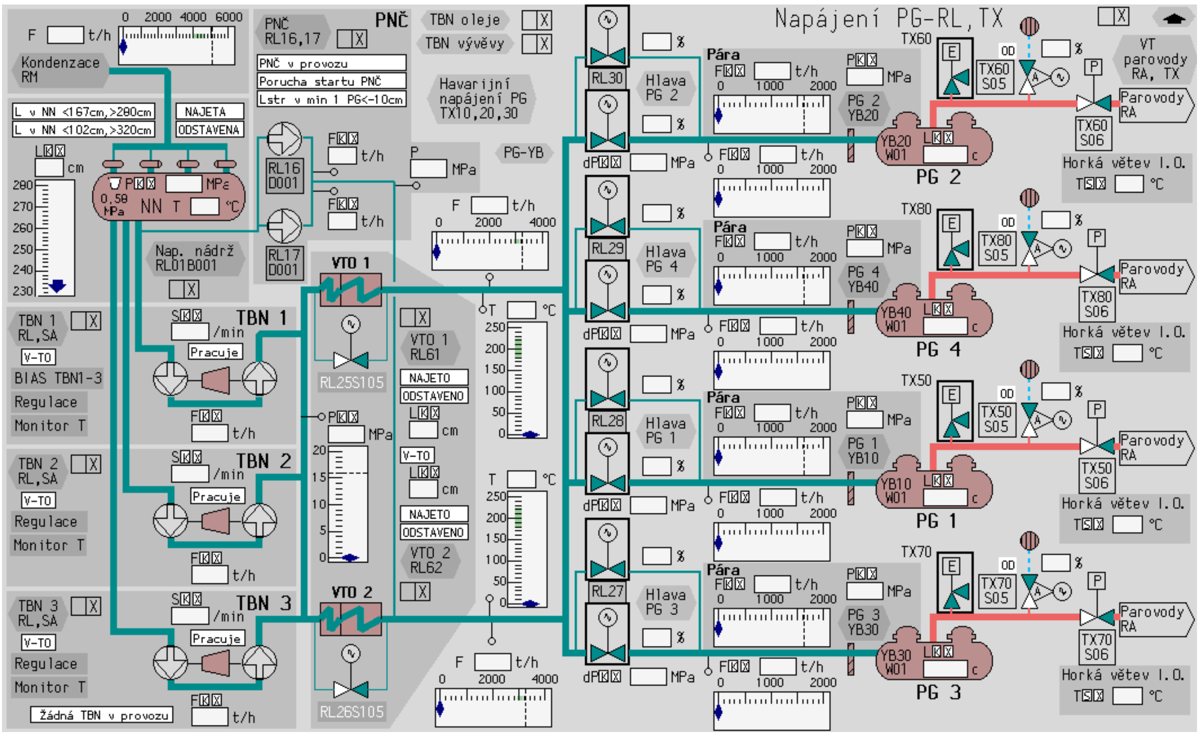


Fig. 1.3.2-3 Auxiliary condenser

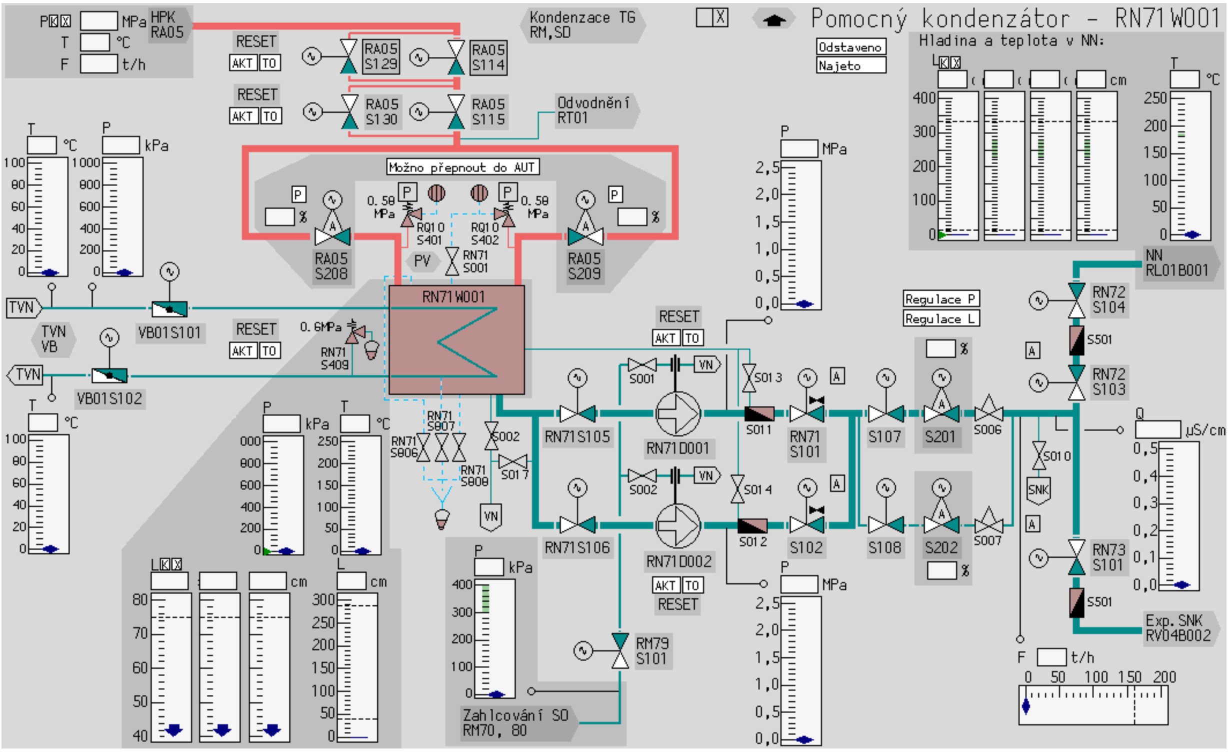
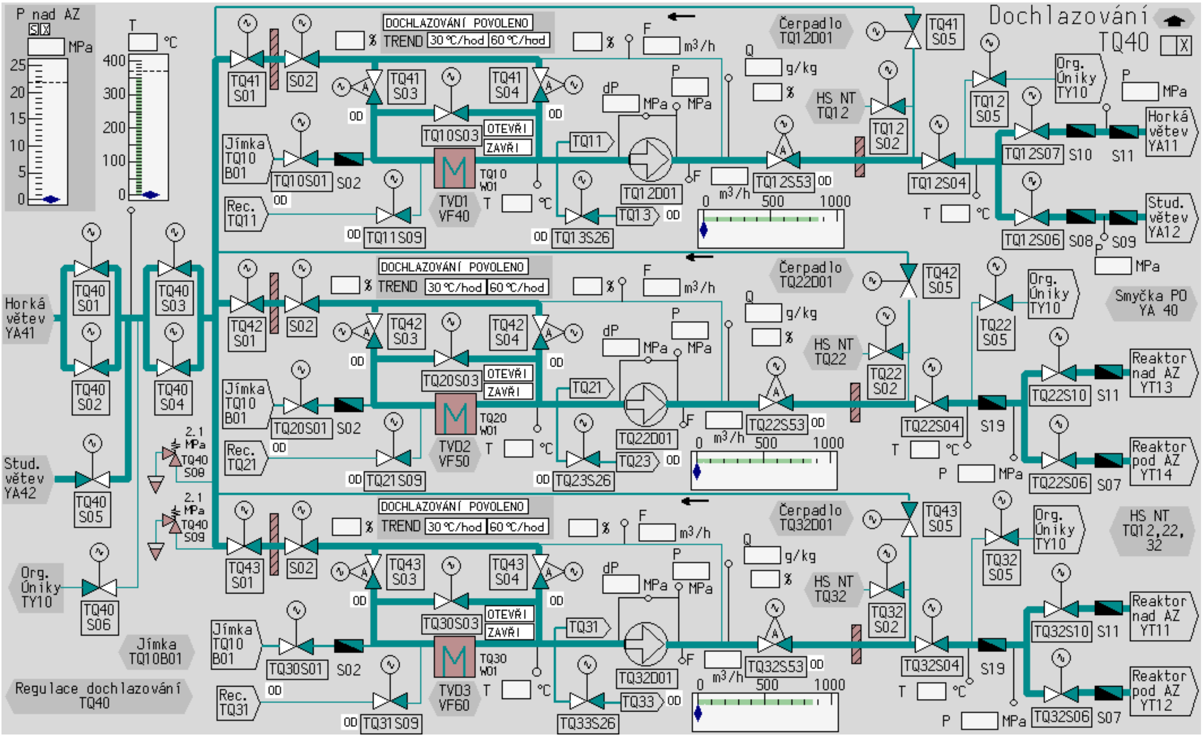


Fig. 1.3.2-4 Residual heat removal system



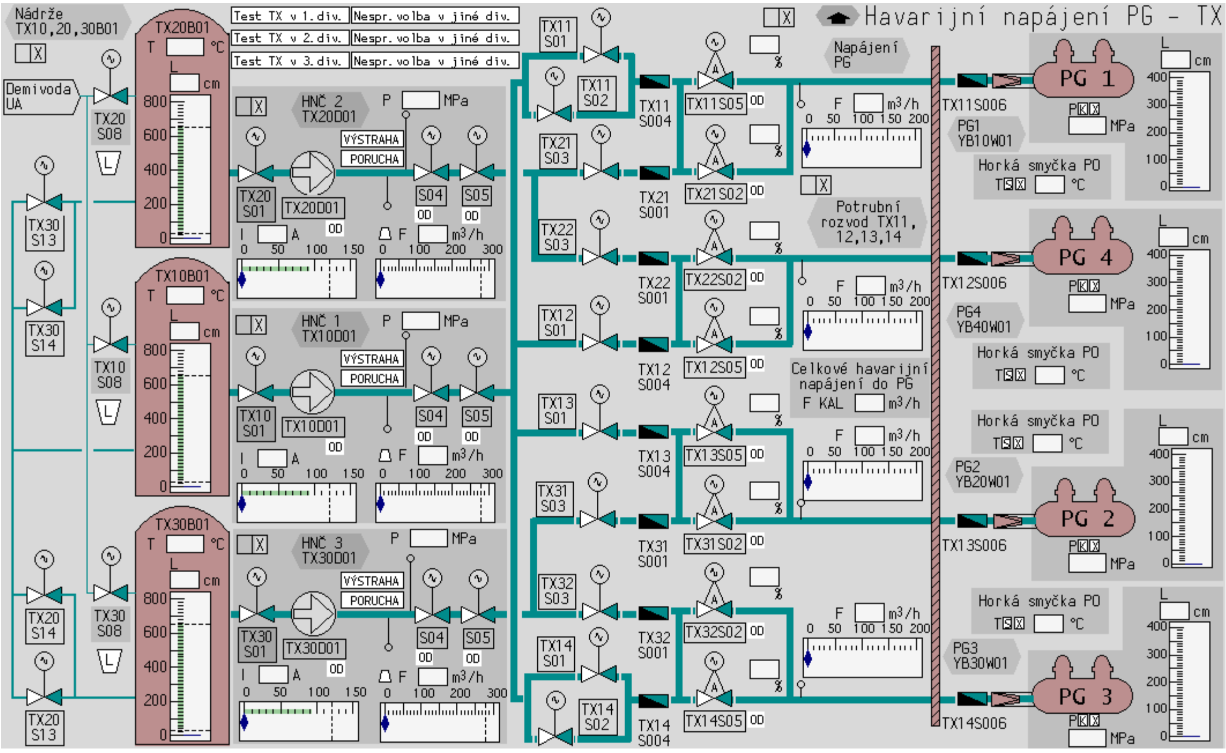
Safety systems

Emergency feedwater system provides SG feeding in case of excessive level drop in two SGs. The system consists of three independent trains (with 3x100 % redundancy), while each train includes an emergency feedwater pump, demi-water storage tank with the capacity of 500 m^3 and piping lines. All three tanks are interconnected by two couples of isolating valves in order to provide for possibility to use water from the tanks.

The demi-water storage tanks, including their interconnection, are situated in a separate room. Emergency feedwater pumps and emergency feed water system valves are situated in the reactor building outside containment in separated rooms.

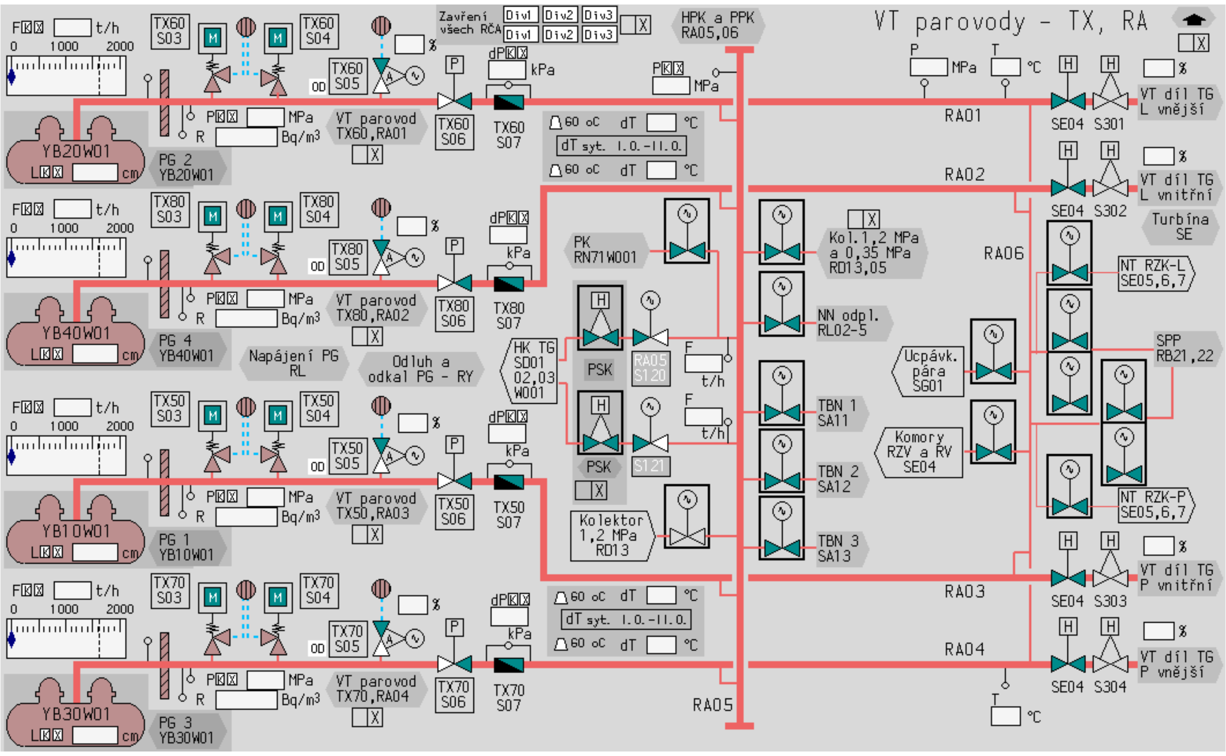
The steam dumps to atmosphere provide the safety function of heat removal from SG by steam discharging to the atmosphere during accidents without SG leakage in case it is not possible to use the steam dump to condenser, i.e. in case of loss of power or when the capacity of the steam dump to condenser is not sufficient to reduce the pressure and prevent pulsing operation of safety vales in all operating modes. The steam dumps are connected to various sections of secured electric power supply from accumulators. The steam dump opens when the set pressure limit is exceeded and SG pressure is controlled by steam relieve. After pressure reduction the steam dump closes.

Fig. 1.3.2-5 SG emergency feedwater system



The next stage of secondary circuit protection against overpressure is by two pulse **SG safety valves**. SG safety valves open gradually in case higher pressure value is exceeded than the opening pressure of the steam dump to atmosphere.

Fig. 1.3.2-6 Main steam lines



The steam dump to atmosphere and SG relieve valves are situated in the reactor building outside containment.

HP emergency core cooling system provides mitigation of the course and elimination of consequences of accidents with primary circuit leakage or possibly secondary circuit leakage. During normal operation of the unit at nominal or reduced power these systems are in standby mode, ready to act in case of emergency. For heat removal from the core in emergency situation it provides:

- RCS injection and boric acid (H_3BO_3) concentration increase in the primary circuit in case of RCS leakage or secondary circuit rupture with the objective to limit fuel damage.
- Together with other safety systems, its function limits leakage of radioactive substances and ionizing radiation from the containment under emergency conditions and after that.

The HP system is designed with 3x100 % system redundancy, including all supporting systems (cooling, power supply, control and ventilation). The HP emergency core cooling system consists of three centrifugal pumps. Each pump has its own tank with a inventory of concentrated boric acid solution (40 g/kg). All pumps are supplied from the essential power supply system (DG).

The system is situated in the reactor building outside of the containment and the delivery pipes pass through to the containment boundary and are connected to the primary circuit. The pumps are installed in independent, physically separated rooms of the safety systems. The HP emergency injection system tanks are situated inside the containment in separated compartments.

In addition, in order to provide for core long term cooling, other inventory of boric acid is available in the containment sump (app. 12 g/kg), common for HP, LP ECCS and the containment spray system.

LP emergency core cooling system provides (in addition to RHR) for mitigation of the course and elimination of consequences of accidents with extensive RCS leakage (large LOCA). During normal operation of the unit at nominal or reduced power these systems are in standby mode, ready to act in case of emergency. For heat removal from the core in emergency situation it provides:

- Primary circuit injection and boric acid concentration increase in the RCS in case of more extensive RCS leakage, and residual heat removal from RCS with the objective to prevent fuel damage.
- Maintains the reactor in safe shut down conditions after each shut down.

- Together with other safety systems, its function limits leakage of radioactive substances and ionizing radiation from the containment under emergency conditions and after that.

The HP system is designed with 3x100 % system redundancy, including all supporting systems (cooling, power supply, control and ventilation). The HP emergency core cooling system consists of three centrifugal pumps. The inventory of boric acid is available in the containment sump, common for all 3 divisions of the safety systems (HP, LP ECCS and containment spray system). All pumps are supplied from the essential power supply system (DG). Coolant for primary circuit injection is cooled in the ECCS heat exchanger by ECW in the emergency modes as well as in the planned RHR mode.

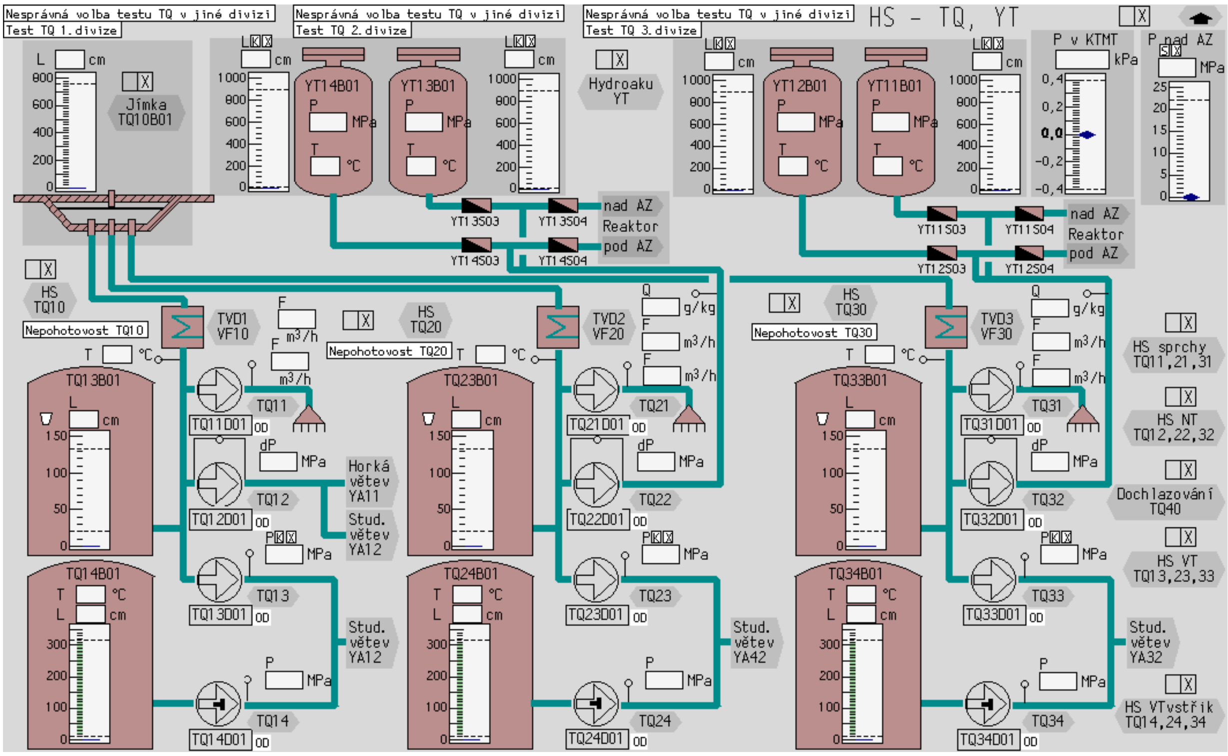
The system is situated in the reactor building outside of the containment and the delivery pipes pass through to the containment boundary and are connected to the primary circuit. Pumps of the LP system are situated in separate, independent rooms of the safety systems, the containment sump presents a separate compartment of the containment with three inlets from the lowest level of the containment, ECCS heat exchangers are situated in three separate, independent rooms outside of the containment.

Passive emergency core cooling system (hydro-accumulator system) serves for quick flooding of the core in case of pressure drop in the RCS in emergency situations associated with pressure drop in the primary circuit, during accidents associated with extensive coolant leakage from the reactor. The system includes four hydro-accumulators providing supply of H_3BO_3 solution (12 g/kg) both to upper plenum and down comer of the reactor. It is a passive system which does not require power supply for its operation. The system consists of emergency core cooling pressure tanks and connecting pipes. Coolant discharge to the reactor is driven by pressurized nitrogen expansion. This system works as a passive system, i.e. in case of pressure drop in the primary circuit under the pressure in the tanks, discharge to the primary circuit is initiated. It is activated without any external initiation impulse and no electric power supply is required. To prevent nitrogen intrusion to the primary circuit after the pressure tanks are emptied there are stop valves installed on the route, supplied from the essential power supply system (accumulator batteries).

The nuclear safety design concept is based on the assumption of possible functional failure of some of the passive system component and therefore redundancy of this system follows the 100+100 % principle. It consists of two independent subsystems of identical function, each with two pressure tanks.

The passive core cooling system is situated inside the containment. The pressure tanks (hydro-accumulators) are installed in couples in separated compartments of the containment.

Fig. 1.3.2-7 Active and passive ECCS and containment spray system



Other systems

In case of a beyond design, highly improbable situation of total loss of heat removal from the core (simultaneous loss of secondary heat removal and primary "feed and bleed") additional strategies are implemented to provide secondary heat removal, using the components beyond their design specification.

SG make up by condensate pumps by bypass of the feedwater tank (the route designed for secondary circuit flushing). To enable condensate flow to SG it is necessary to reduce pressure in at least one intact SG by steam dump to the atmosphere, to the value close to the atmospheric pressure with regard to the shut of head of condensate pumps. This strategy is described in EOPs.

Other possible use of components beyond their design specification is **gravitation SG feeding from the feedwater tank**. Since the feedwater tank is situated on the elevation of +30.0 m and SG on the +28.8 m, it is necessary to reduce pressure in at least one undamaged SG by steam dump to the atmosphere, to the value close to the atmospheric pressure, to enable gravitation SG feeding. Operating pressure in the feedwater tank is app. 0.6 MPa even with shut down reactor and therefore it provides the condition for at least partial heat removal from the primary circuit by gravitation SG feeding from the feedwater tank. This strategy is described in SAMG.

1.3.2.3 Time constraints for availability of different heat transfer chains

Based on engineering judgment, trivial calculations and comparison with existing analysis it is possible to identify minimum time for which individual systems are capable to remove heat from the core.

Operating systems

The auxiliary SG feedwater system feeds water to SG from the feedwater tank. The feedwater tank capacity is 350 m^3 of water. In addition it is possible to make up the feedwater tank from turbine condenser with the capacity of app. 250 m^3 of water. It is also possible to make up the feedwater tank from demi-water storage tanks with the capacity of 2 x 770 m^3, which are common to both units. This volume of water available for SG feeding is sufficient for cooling the unit down to parameters which allow initiation of residual heat removal system operation, followed by unit cooldown to cold shutdown conditions.

When RCS parameters are decreased (RCS temperature below 150 °C and RCS pressure below 1.7 MPa) the residual heat removal system is activated to continue cooldown to the cold shutdown state in a closed circuit (core – ECCS exchanger – LP ECCS pump - core). In this RHR mode it is necessary to provide RCS make-up only with coolant necessary to compensate for temperature volume changes due to coolant shrinkage. For this purpose there is sufficient coolant inventory with corresponding boric acid concentration in the storage tanks of both operating and safety systems and means for coolant make-up to the primary circuit are available. The unit may be maintained in the cold shutdown state in the closed circuit of residual heat removal for unlimited period of time.

There are no time limits set for use of operating systems for residual heat removal and maintaining the unit in cold state.

Safety systems

When the emergency SG feedwater system is in use water is fed to SG from the storage tanks of this system and heat is removed by steam dump from SG to atmosphere (secondary "feed and bleed"). When demi-water is fed to SG, storage tanks 3 x 500 m^3 of the emergency feed-water system are available for each unit, plus the tanks 2 x 770 m^3, common to both units. This water inventory easily suffices for unit cool-down into a cold shutdown state (the design basis specifies one emergency SG feed-water system as sufficient for cooling down the unit into a cold shutdown state) or for maintaining the units in a hot state for approximately 72 hours.

If for whatever reason the secondary circuit is unavailable for heat removal from the core, emergency core cooling systems in the primary "feed and bleed" mode can be used in hot or semi-hot state. In this mode controlled coolant discharge is conducted from the RCS (by RCS emergency venting system or PRZR relief valve) through bubbler tank to the containment, and the coolant is than injected back to RCS by the HP ECCS pump through ECCS heat exchanger (from which the heat is removed to the ultimate heat sink by ECW system). In this case heat released to the containment is removed by containment spray system also through ECCS heat exchanger to ECW.

There are no time limits set for heat removal using the primary "feed and bleed" system.

Other systems

For SG feeding using condensate pumps an inventory of 2x 770 m^3 demi-water is available common to both units. If water feeding to SG by condensate pumps is effective, this inventory of water provides sufficient capacity for unit cool down to the cold shutdown state and unit maintenance in this state for at least 24 hours, which is sufficient for implementation of additional actions in order to provide heat removal from the core using different means.

In case of gravitation SG feeding from the feedwater tank there is 350 m^3 of water available which is normally stored in the feeding tank. Considering the fact that this is an ultimate solution in case of total loss of all means of heat removal from the core, this water inventory provides time to implement other activities leading to heat removal from the core by other means.

1.3.2.4 AC power sources and batteries

Power supply of all core heat removal systems is implemented within the following protection in defense-in-depth hierarchy:

1) from main power supply sources (from home consumption transformers) or
2) from auxiliary power supply from the grid, or
3) from the emergency power supply from safety and common DGs and batteries

Operating systems which provide primary circuit sub-criticality as well as secondary circuit heat removal systems are supplied from the safety related power supply systems (common DGs and corresponding batteries). All safety systems are supplied from the secured power supply system of safety systems (DGs and batteries). After loss of off-site power supply and connection of corresponding DG the system pumps are activated by the automatic loading sequencer.

Pumps and the related valves, which require change of their position to fulfill safety functions, are supplied from the respective DG. Selected key components (valves of the steam dump to atmosphere, isolating valves of hydro-accumulators, quick acting containment isolation valves) and related I&C systems are supplied from batteries. For details concerning power supply see the sections 1.3.5 and 1.3.6.

1.3.2.5 Need and method of cooling equipment

Operating systems

Unit cooling by operating systems takes place in two phases. During the first phase residual heat from the core and accumulated heat is removed through the secondary circuit by means of steam dump to atmosphere either by direct steam dump from SG to atmosphere (until RCS temperature is below 110 °C) or steam dump to turbine condenser (until RCS temperature is below 100 °C)and from there to cooling towers by cooling water. During the second phase residual heat from the core and accumulated heat is removed using RHR system via ECCS heat exchangers by ECW and ECW cooling pool to atmosphere as the ultimate heat sink.

The steam and water modes can be considered as two diverse methods of cooldown, since they use different systems for their operation. However due to the physical principles the unit can not be cooled down to the cold status in the steam mode and RHR system could be used after RCS temperature is decreased below 150 °C.

Safety systems

The safety systems of secondary heat removal (feeding of SG using the emergency feed water system and steam dump from SG) provide heat removal directly to atmosphere during the steam mode. The ECW system is required for heat removal from safety system components (pumps, I&C, MCR/ECR ...) and also from DG in case of loss of off-site power supply. However heat removal from the secondary side is the simplest and feasible with minimum means.

In case the secondary heat removal is unavailable or during the second phase of cooldown by RHR system, the residual heat from the core and accumulated heat is removed via ECCS heat exchangers by ECW and ECW cooling pool to atmosphere as the ultimate heat sink.

ECW system operation is required also for primary bleed and feed, since heat is removed through ECCS heat exchangers by ECW and ECW cooling pool to atmosphere as the ultimate heat sink.

Other systems

During SG feeding by condensate pumps or during gravitation feeding of SG by water from feedwater tank evaporated water from SG is dumped to atmosphere. Basically it is a passive method of heat removal to atmosphere, since if sufficient amount of water is fed to SG for heat removal from the RCS, steam is dumped to atmosphere by simple remote opening of the steam dump valve (supplied from accumulator) or by manual opening locally (the valve is located outside of the containment at accessible place in the reactor building).

1.3.3 Heat transfer from spent fuel pools to the ultimate heat sink

Spent fuel pools (SFP) are situated in the containment and consist of two sections. Two possible initial conditions exist for residual heat removal from SFP:

- SFP contains spent fuel from previous fuel cycles in order to achieve reduction of its activity and generated residual heat.
- SFP contains all fuel from of the core, when there is partly spent fuel with high residual heat together with spent fuel from previous fuel cycles.

During SFP operation in the fuel storage mode the water level in the pool is maintained higher than 792 cm, which is sufficient for both shielding and cooling functions. Volume of water in SFP in the fuel storage mode is 223 m^3 in each sections 01 and 03, and 104 m^3 in the section 02 (the volume of approximately double in the refueling mode). In both cases heat removal is provided by forced water circulation in SFP provided by dedicated SFP cooling system (3x100 %) through heat exchangers cooled by ECW system, which removes heat through ECW cooling pool to atmosphere as the ultimate heat sink.

1.3.3.1 Existing heat transfer means

For heat removal from SFP operating systems (classified as safety related systems) can be used in combination with safety systems, but also systems classified within safety systems category only.

<u>Operating systems of heat removal from SFP</u>

The normal operating system for heat removal from SFP to the ultimate heat sink is the SFP cooling system, which provides heat removal from SFP through heat exchangers from where heat is removed by ECW (3x100 %). ECW systems transfer the heat to the atmosphere in the spray cooling tanks. In this mode flow of the cooled coolant from heat exchangers back to SFP is provided by SFP cooling pumps.

<u>Safety systems</u>

In case it is not possible to remove residual heat from fuel in SFP, residual heat can be removed by water injection to SFP by any of the three containment spray pumps. The source of the cooling water is the containment sump and coolant from SFP is discharged by overflow to the containment, returned to containment sump and back to SFP by the spray pump through ECCS heat exchanger (from where the heat is removed by ECW system to the atmosphere as an ultimate heat sink).

Other systems

In case any of the pump-exchanger-ECW combinations are not capable to remove residual heat from fuel in SFP, it is possible to remove residual heat by evaporation of injected coolant into the containment, provided that the source of cooling medium is the containment sump and coolant is injected to SFP by any of the three containment spray pumps.

Other possible method of SFP make up is use of the SFP fuel treatment system pump.

1.3.3.2 Layout information on the heat transfer means

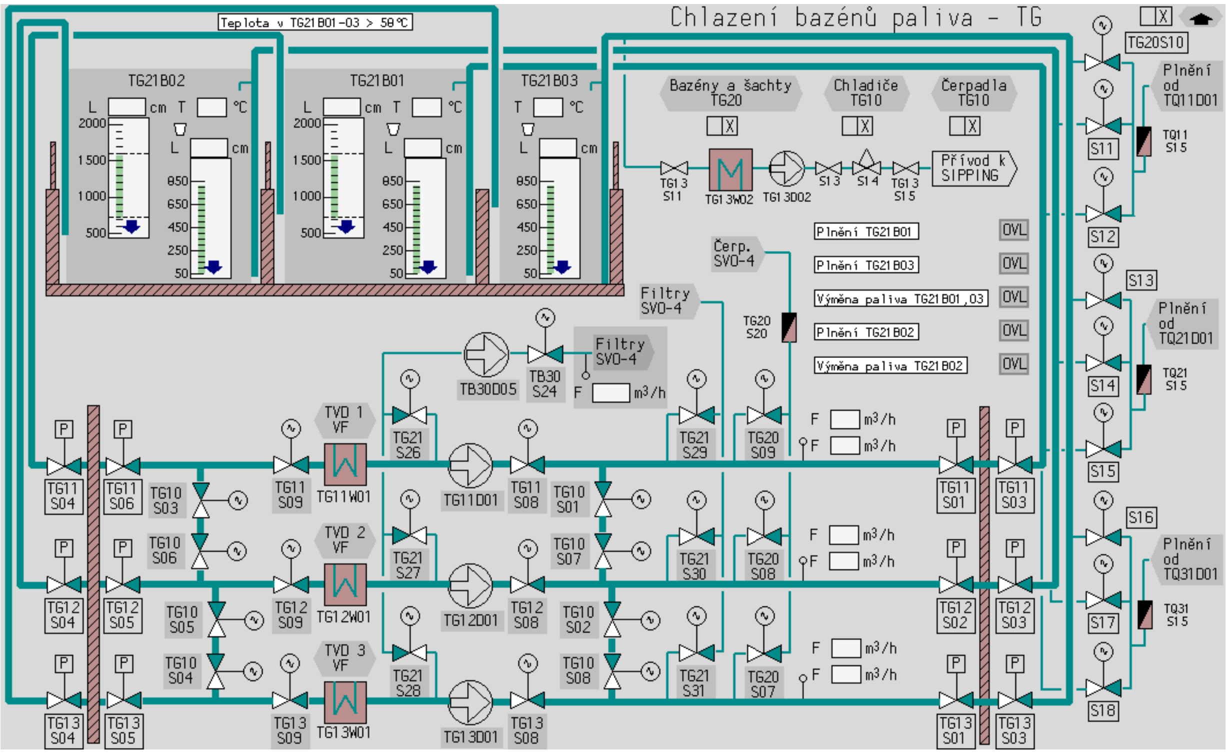
Operating systems of heat removal from SFP

Under normal operating conditions heat removal is provided by one of three SFP cooling system circuits, classified as safety related system with 3x100 % redundancy. The coolant is boric acid solution, although theoretically SFP can be filled with pure condensate as well. The cooling circuit is of seismic resistance. Cooling circuit capacity is sufficient for both initial conditions of SFP content.

Coolant circulation is provided by three separate cooling circuits for individual sections of SFP. Each section includes a pump and an heat exchanger cooled by ECW. To increase reliability and availability of the entire SFP heat removal system the said three circuits are interconnected on the pump suction side and also on pump discharge side providing operative combinations of heat removal chain (SFP section filled with spent fuel, pump, heat exchanger with particular ECW system). Pumps are supplied from the essential power supply system of the safety systems (DG). SFP cooling system is situated outside of the containment, SFP itself is situated inside the containment.

Considering the fact that it is a closed cooling circuit on the SFP coolant side and SFP is covered, there is only very small evaporation. Therefore operating systems of SFP cooling provide heat removal from spent fuel placed in SFP for unlimited time.

Fig. 1.3.3-1 SFP cooling system



Safety systems

In case of a total loss of normal SFP cooling (either due to water level drop or after interruption of heat removal), the containment spraying system shall be used for make-up to SFP, in alignment for emergency make-up to SFP. With the use of this system, it is possible to make-up SFP and to ensure the drainage of coolant from SFP, by means of an overflow to the containment sump, which ensures heat removal from the spent fuel in SFP in an alternative way through the ECCS heat exchanger. This cooling circuit is independent of the normal SFP cooling system and provides an alternative way of heat removal from the spent fuel stored in SFP. However, even during emergency cooling, heat from spent fuel in SFP, via ECCS heat exchanger, is removed to ECW.

If it is possible to close the cooling circuit via the containment sump and ECCS heat exchanger, than this SFP cooling method also provides heat removal from spent fuel in SFP for unlimited period of time.

Other systems

In case any combination of forced circulation circuits (pump, heat exchanger, ECW) is not capable to provide heat removal from SFP, than heat is removed by coolant boiling and evaporation from SFP to containment. The source of water for SFP make up in this mode is the spray system (classified as safety system) which is capable to supply water through a side branch from the discharge line of any of the three spray pumps from the containment sump to any of the three SFP cooling circuits and therefore also to any of the three SFP sections. The capacity of the containment sump is app. 580 m^3 of coolant.

Other possible method of SFP make up is use of SFP coolant treatment system pump which may make-up coolant to SFP from coolant storage tanks dedicated for refueling. These pumps are supplied only from non essential power supply sources. The capacity of the storage tanks is app. 1,600 m^3 of coolant. While these tanks are common to both units the volume of coolant for SFP make up in the affected unit may be limited in case refueling is ongoing in the other unit.

With unsealed reactor and open gate between SFP and refueling pool there is a possibility of coolant injection by any HP or LP ECCS pump to the primary circuit and from there to SFP, or coolant injection from hydro-accumulators can be provided.

1.3.4 Heat transfer from the containment to the ultimate heat sink

The design function of containment is limitation of potential radiological consequences in case of reactor plant accident. This function is, among others, provided by the containment structure, which limits leakage outside the containment to very low values even at a high internal overpressure inside the containment. Since the whole pressure interface of the RCS is located inside the containment, the containment acts as the last barrier against the leakage of radionuclides that can be released from the fuel or RCS coolant in case of an accident.

The integrity of containment is ensured in the design using the following safety systems:

- Containment isolation system – isolation valves automatically closed in case of pressure increase inside the containment.
- Containment pressure reduction system - spray pumps and inventory tanks with chemical agents to trap post-accident iodine.
- Post-accident hydrogen disposal system - passive auto-catalytic recombiners, designed for design-basis accidents.

Calculated design-based containment load:

Maximum temperature	150 ^oC
Maximum pressure	0.49 MPa
Dose rate	10^3 Gy/hour

Ventilation systems installed inside the containment are intended for containment cooling under normal operating conditions. In case of accident with pressure in the containment exceeding 0.3 kPa the ventilation systems are disconnected and stop to operate. In addition valves of ventilation systems, which maintain under pressure in the containment, are closed. Heat removal from the containment and pressure reduction under emergency conditions is provided by spray system in combination with other safety systems.

1.3.4.1 Existing heat transfer means

For heat removal from the containment operating systems (classified as non essential systems or safety related systems) in combination with safety systems can be used, but also systems classified within safety systems category can be used only.

<u>Operating systems of heat removal from the containment</u>

Normal operating systems of heat removal from containment to the ultimate heat sink are the containment ventilation circulating systems. Heat removal from the containment is provided in heat exchangers, from where heat is removed by ECW. ECW transfers the heat to the atmosphere in the spray cooling tanks. Alternatively heat exchangers can be used for heat removal, from where heat is removed by chilled water.

<u>Safety systems</u>

The spray system provides heat removal from the containment in case of accidents when overpressure in the containment (in case of primary or secondary coolant leakage inside containment) exceeds 30 kPa. The purpose of the spray system is to provide condensation of part of the coolant and so reduce pressure inside the containment and eliminate release of fission products to environment. This is achieved by spraying of cold water with boric acid, followed by steam condensation. Heat removal from the containment is provided by the spray system, provided that water flows to the containment sump, passes ECCS heat exchanger where heat is removed to ECW and than water returns back to the containment.

<u>Other systems</u>

In case of unavailability of heat removal from the containment by ECW system with the use of the spray system it is possible to partly provide heat removal by alternative water spraying in the containment. Fire extinguishing system for RCP can be used for this purpose. With this system it is possible to spray extinguishing water into the containment with similar effect as normal containment spray system.

1.3.4.2 Layout information on the heat transfer means

Operating systems of heat removal from the containment

Under normal operating conditions heat removal from the containment is provided by containment ventilation circulation systems. These systems are classified as safety related systems.

SG compartments circulation cooling system provides removal of heat and steam transferred to air by technology in the containment, in order to maintain air temperature within required limits. Heat and steam is removed on surface coolers. Warm air is exhausted from the rooms and upon treatment (cooling) returned back to individual compartments.

The system is designed with 2x100 % redundancy (3 operating + 3 back-up fans), there are two serial coolers allocated to each fan. One cooler is connected to ECW distribution and the other cooler to chilled water. Fans are supplied from the essential power supply system of the safety systems (DG). Fans and coolers are installed inside the containment.

Reactor cavity circulation cooling system provides reactor cavity cooling. Air is withdrawn from SG compartments, cooled on surface coolers and so treated air is delivered by pipes to the reactor cavity from where it is withdrawn back to SG compartments.

The system is designed with 3x100 % redundancy, i.e. three fans, from than one operating and 2 back-up fans. Coolers are connected to ECW distribution system and the chilled water system. Fans are supplied from the essential power supply system of the safety systems (DG). Fans and coolers are installed inside the containment.

Reactor hall cooling system provides cooling of the top part of the containment. Air is withdrawn from SG compartments and after cooling in two-stage cooler it is blown to the reactor hall.

The system is designed with 3x100 % redundancy, i.e. three fans, from than one operating and 2 back-up fans. Fans are supplied from the essential power supply system of the safety systems (DG). The first series of coolers is connected to ECW distribution system and the other to chilled water distribution system. Fans and coolers are installed inside the containment.

Control rods drive mechanisms cooling system provides removal of separated heat from control rods drives. Air is withdrawn from containment compartments and cooled air is blown to SG compartments.

The system is designed with 3x100 % redundancy, i.e. three fans, from than one operating and 2 beck-up fans. Coolers are connected to non-essential service water distribution system. Fans are supplied from the essential power supply system of safety related systems (DG). The system is installed inside the reactor hall in the containment.

Circulation systems with heat exchanger cooled by ECW may remove heat from containment for unlimited period of time.

Safety systems

Containment spray system is designed for maintaining conditions in the containment during emergency situations. Pumps takes suction from the containment sump via ECCS heat exchanger and coolant is sprayed into the containment by the spray nozzles. Steam condenses on drops of water and thus pressure in the containment reduces. Water flows to the containment sump and than heat is removed to the atmosphere by ECW in ECCS heat exchanger.

The spray system is designed with 3x100 % redundancy concept, including all supporting systems (cooling, power supply, control and ventilation); it consists of three technologically and functionally identical but independent subsystems, while each of them is capable to perform independently any tasks for which they have been designed. Each of the 3 subsystems contains a spray pump, a storage tank containing H_3BO_3, N_2H_4, KOH for iodine aerosols retention), water jet pump, ECCS heat exchanger, connecting pipes and spray nozzle system. Pumps are supplied from the essential power supply system of the safety systems (DG).

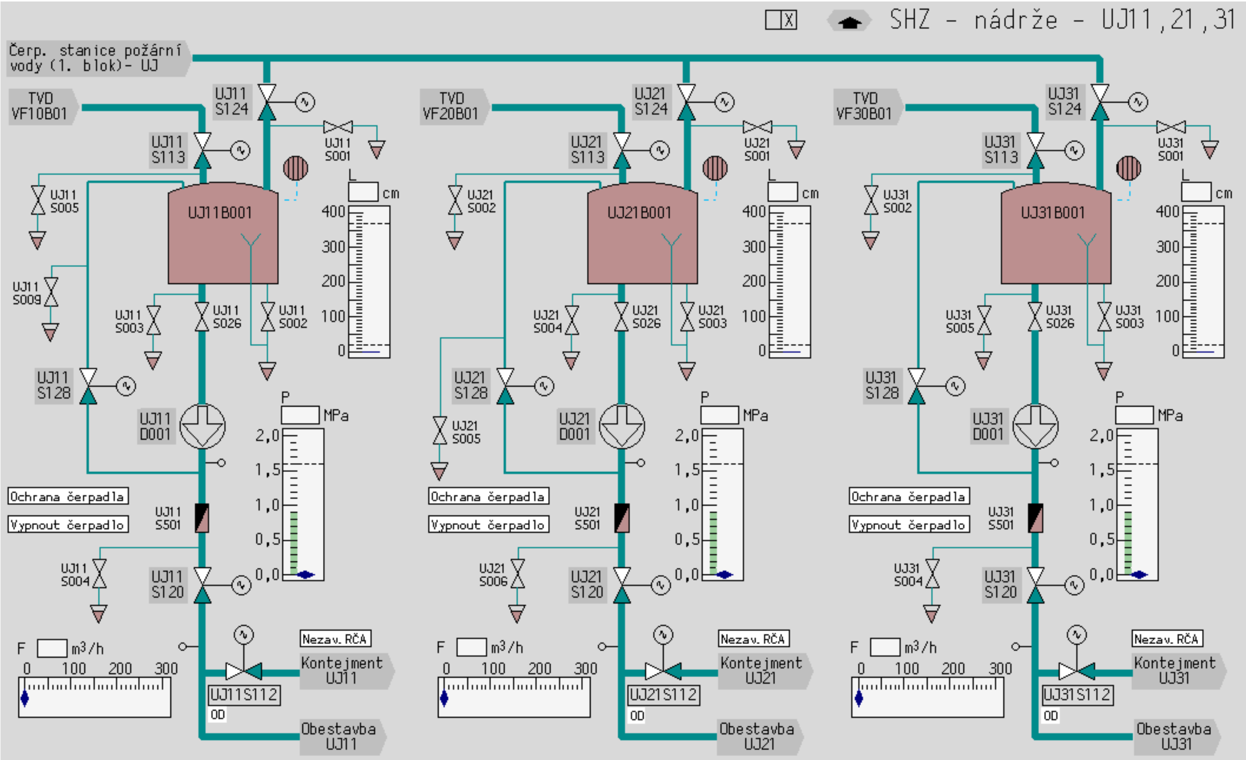
The spray system enables emergency SFP make up if required.

Pressure reduction and heat removal using containment spray system takes place in a closed circuit via containment sump. Therefore heat removal from the containment can be applied for unlimited period of time in this mode.

Other systems

Alternatively heat from the containment can be removed by fire water spraying in RCP motor compartments. Although this system has been designed for extinguishing of potential fire of RCP motors, it can be used with similar effect as the standard containment spraying since the RCP motor compartments are connected with other containment compartments.

Fig. 1.3.4-1 Fire extinguishing system



The fire extinguishing system pumps take suction from storage tanks with the capacity of 3x70 m^3 and spray it into containment via nozzles for RCP motor fire extinguishing.

The fire extinguishing system is designed with the 3x100% redundancy concept, including all supporting systems (cooling, power supply, control and ventilation); it consists of three technologically and functionally identical but independent subsystems. Pumps are supplied from the essential power supply system of the safety systems (DG). The pumps and the storage tanks are situated outside of the containment.

Capacity of the storage tanks is designed for fire extinguishing. For the purpose of containment spraying fire fighting storage tanks can be made-up from the ECW system. This combination provides alternative means for containment spraying for unlimited period of time.

1.3.5 AC power supply

The following "defense-in-depth of electric systems" concept is applied in Temelín NPP electric systems in relation to the nuclear island design.

DID structure and robustness of individual levels result in resistance against both external and internal events (failures).

Tab. 1.3.5-1 Defense-in-depth levels in electric systems

DID level	NPP	Electric systems	Level robustness
1	Prevention of abnormal operation and failure	• Insensitivity to deviations of U,f • Power distribution stability • Dynamic stability • Separated grid operation	• Reference to SSC robustness • Electrical systems obustness (independence, redundancy, diversity) • Protections • Controls, automatics • Quality • Functionality testing • Operating instructions • Trained personnel • etc.
2	Control of abnormal operation and failure	• Load rejection to home consumption • Transfer to back-up power supply	
3	Control of accidents within the design basis	• Design (safety) functions of essential power supply systems • Safety systems (1,2,3) • Safety related systems (4,5)	
4	Control of severe plant conditions that were not explicitly addressed in the design	• Procedures for coping with SBO • Measures for SA consequences mitigation • (AAC functionality)	
5	Mitigation of radilological consequences of significant releases	Support of emergency control centres	

Essential protection and control systems and operating systems with relations to safety functions are supplied from redundant essential power supply systems. Every unit has 3 independent safety supply systems (identified as 1, 2, 3) and additional supply systems classified as safety related (identified as 4 and 5). These systems provide supporting safety functions - essential power supply and participate also in control of electric loads function.

1.3.5.1 Off-site power supply

The NPP Temelín is situated in the southern region of the Czech Republic and consists of two VVER 1000 MWe units. The grid connection as well as home consumption system diagram (operating and back-up) is unit based in accordance with general design concept. The output power is transferred to 400 kV transmission grid. The auxiliary power supply is provided from 110 kV distribution grid.

Connections of the plant with external power grids

The turbine generator of each unit includes 1111 MVA, 24 kV generator. Generator power output is transferred via generator breaker and unit transformer (1200 MVA, 420/24 kV) by a separate single line to 400 kV switchyard in Kočín, which is in app. 3 km distance from Temelín NPP. 400kV breakers of NPP units are installed in that switchyard.

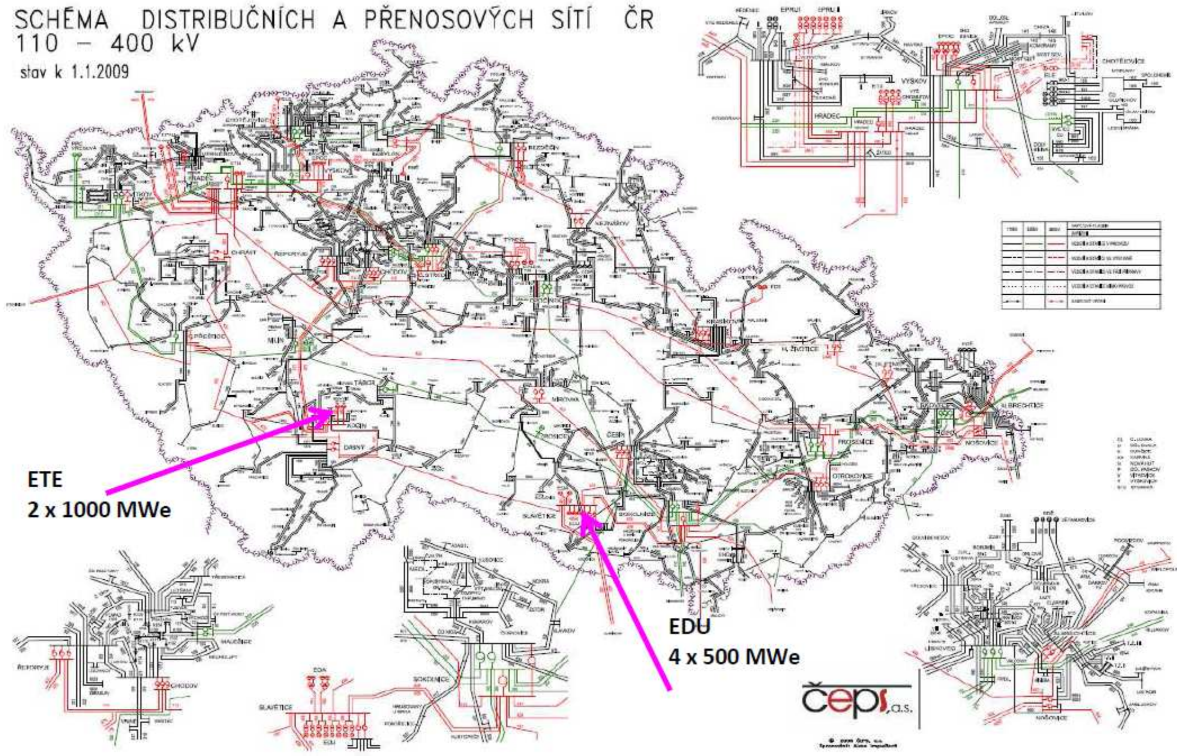
The 400 kV switchyard in Kočín is connected to the transmission grid by 5 lines, which distribute power to various distant parts of the Czech Republic (Central Bohemia, Western Bohemia, Southern Bohemia). This creates geographic diversification of 400 kV connections. Some lines are single others are double. The Czech transmission grid as a whole is designed and operated in accordance with N-1 criteria. Power output from Kočín switchyard further to the grid is however designed in compliance with the stricter N-2 criteria. These requirements are set in the Transmission Grid Codex.

The 400 kV Kočín switchyard is of outdoor type with short circuit resistance of 50/125 kA. Its diagram includes two bus systems and an auxiliary bus. Temelín NPP´s units are connected to that part of the switchyard, which is designed with higher requirements for reliability and resistance of NPP units connection against grid failure. The diagram of this part of the switchyard includes 4/3 breakers to the branch with 4 out of 5 400 kV lines connected to it, connecting the switchyard to the transmission grid. Other part of the switchyard has a conventional diagram with one breaker per branch. This is the place of connection of the 5th 400 kV line, including two 400/110 kV transformers, 250 MVA for power supply of 110 kV Kočín switchyard and a damper which contributes to control of idle power balance in this node.

The 110kV Kočín switchyard serves as the main source of back-up power supply of NPP´s unit in particular. It is also a source of power supply for NPP´s water pumping station situated at nearby Hněvkovice dam on Vltava river. In addition the switchyard supplies the 110 kV distribution network in Southern Bohemia. The 110kV Kočín switchyard has a robust and flexible diagram with 3 bus systems.

Diagrams of both 400 kV and 110 kV Kočín switchyard and the method of their operation are selected to eliminate transmission of failures between NPP´s units as well as between NPP´s units and the electric power grid.

Fig. 1.3.5-2 NPP incorporation in the Czech Republic's grid



Kočín cooperates in parallel with app. 30 km distant switchyard 400/110 kV in Dasný, with 2 transformers 400/110kV of the same capacity as in Kočín. Kočín and Dasný switchyards are connected in parallel with two lines in both 400 and 110 kV parts. This connection provides back-up of 400/110 kV transformers, while maintaining sufficient voltage for back-up power supply for Temelín NPP in Kočín 110 kV switchyard.

Kočín 110kV switchyard therefore may be supplied from different geographically and directionally diverse sources in the transmission grid (400/110 kV transformation in Kočín and Dasný, 220/110 kV switchyard in Tábor) as well as in 110 kV distribution grid (Lipno hydro power plant).

NPP units are capable to operate within separated section of the power grid, i.e. in so called island operation.

Turbine control system includes also a specific separated grid controller (proportional frequency control) with the primary function to maintain frequency in the separated grid. The turbine "island" mode is activated by grid frequency deviation. For scenarios associated with extensive turbine load rejection the design assumes a function of high turbine acceleration evaluation and the derived impulse to turbine hydraulic control accelerator (so called runout control). At the same time turbine "island" mode is the input for main unit control system in order to provide necessary coordination of reactor, turbine and steam dump to condenser control modes. A specific procedure exists for the island operation within the operating procedures intended for abnormal operation.

The range of separated grid frequency at which the units are capable to operate on long term basis, is limited by frequency relay setting (the unit is automatically disconnected from the separated grid and load is rejected to home consumption when frequency is less than 47.9 Hz or greater than 51.5 Hz).

The capability of separated grid operation control at both units has been successfully tested during start up of units in 2001 – 2003 years. The tests proved high quality of turbine speed control by separated grid controller as well as other functionalities supporting operation in the island mode. In addition the NPP operated successfully in real slightly excessive island during UCTE grid decomposition to three insulated unit on 4 November 2006.

Capability of separated grid operation of NPP is certified by a certification body as a supporting service for Czech transmission grid operators.

<u>Information on reliability of off-site power supply</u>

During operation of Temelín NPP (from initiation of commercial operation until now) no failure in 400 kV and 110 kV grids has been detected, which would indicate unsatisfactory function or reliability of off-site power system or incorrect response of NPP to off-site grid failures.

During commissioning of the units and during their operation several failures or spurious actuation of electric protection systems occurred. Such events are associated with commissioning of the units and tuning up of individual systems in particular. The following failures can be listed:

Tab. 1.3.5-3 Significant failures of electric systems

01/2002	Slipping of Unit 1 generator due to incorrect function of exciting system during commissioning tests. It was caused by incorrect function and setting of exciting system. The cause was identified and corrected.
05/2004	Loss of main power supply of unit 2 at switch on of the unit 1 transformer. On the unit 2 the automatic switch over to back-up power supply worked correctly. The cause was subject to analysis and it was identified as an excess stroke of magnetizing current at unit transformer start up. Solved by modification of the system of preventing electric protections sensitive to magnetic stroke.

2004 - 2010	Several defects of electric systems (chains of 400 kV line isolators, spurious actuation of flash protections, incorrect action of transformer technological protections after periodical maintenance etc.). Causes of these failures have been identified and corrected.

The above failures, which occurred during commissioning of the units and trial operation, were subject to detail analysis and corresponding corrective measured were adopted and implemented. As a result of that the Temelin NPP units proved their high resistance to grid failures during extensive failure in UCTE grid in November 2006.

2006	Temelín NPP´s units coped successfully with extensive system failures (decomposition of UCTE grid in 11/2006, major failure in the Czech transmission grid in 08/2006) and contributed to stabilization of situation in the transmission grid due to their resistance to voltage and frequency deviations and their control capabilities. After a short circuit in Sokolnice switchyard station on 3 August 2011 the unit 2 successfully run out to the "Island" mode. Reactor output was reduced from 970 MW to 400 MW and consequently, within one minute, the island was reduced to 206 MW (after app. 2.25 mins after run out to the island) and the turbine unit speed changed quickly and reached the max. value of 3019 s^{-1}. The "Island" mode ended after about 50 minutes and the unit started normal operation. During a major failure in UCTE grid on 4 November 2006 the unit 1 of successfully run out to the "Island" mode. Reactor output was reduced from 975 MW to 820 MW and turbine unit speed changed quickly and reached the max. value of 3029 s^{-1}. The "Island" mode ended after about 70 minutes and the unit started normal operation.

Reliability of Temelin NPP connection to off-site networks and resistance to failures are based on the following conditions:

- Unit based diagram of connection to the grid. It limits transmission and expansion of failures between units. In combination with the robust diagram of Kočín switchyard (4/3 switches per branch, sectional division of buses, selective protection system) it limits also transmission of failures between the units and the transmission grid. NPP power output to the grid is designed according to the N-2 reliability criteria.
- Great functional and physical independence of the 400 kV power output system (i.e. main off-site power supply) and the auxiliary off-site power supply system (110 kV). Possibility

to supply the auxiliary power supply system from geographically and functionally diverse sources.

- Responses of the units to failures and transients in the off-site grid are controlled by set of controllers, automatics and protections. These functions are coordinated to provide inter-selectivity and to follow individual level of defense-in-depth protection of the unit in a controlled manner when required.
- Static stability of power output transmission to the grid. NPP units are normally used for automatic secondary control of voltage and idle power. This system ensures stable voltage in the Kočín pilot node and controls positions of limit protection system of generator under-exciting in accordance with external impedance of the off-site transmission grid.
- Stability of turbine unit during short circuits in the power output system (quick basic and backup protections which switch off the short circuit, effective turbine control and generator voltage control, quick control of turbine valves). Stability was subject to analysis in the transmission grid dynamic model. The turbine units are naturally stable due to action of the basic and backup protections (up to 100 ms). In case of longer short circuit duration (action of 400 kV breaker failure automatic) the function of turbine valve quick control contributes to turbine stability.
- Capability of units to operate in island mode of the transmission grid, associated with high deviations in frequency and voltage (support of grid stability during system failures). The units may operate at the full power output within the range of 49÷50,5 Hz. Time and power limited unit operation is possible within the range of 47.5÷51 Hz. Units are provided with grid frequency protections, which switch over unit power control to "island mode" at the 1st stage (±200 mHz). This is the way the unit controls its power output in order to stabilize U and f conditions in the island grid. If the situation is not stabilized successfully and the frequency deviation rises and exceeds the limit (47.9 Hz for 1s, or 51.5 Hz for 10s) the 2nd stage of frequency protection switches off the unit from grid and rejects power to home consumption.

  Reactor protection system measures RCP input power and therefore is sensitive to frequency and rate of frequency drop in the grid. In case of quick frequency drop in the grid (major power instability) it is preferred to shut down the reactor by the protection before turbine power rejection to home consumption by turbine control system. In this case home consumption supply of the units is switched over either to auxiliary power supply from 110 kV grid (automatic backup) or essential power supply systems are disconnected from the grid and switched over to emergency DGs.

- The main unit switchyards are provided with automatic switch over from main to auxiliary power supply (110 kV). Both quick and backup under voltage channels of auxiliary power supply are available. In case of loss of main off-site power supply (e.g. by action of generator protection, unit transformers or other power output components, or in case of unsuccessful turbine run out control) the switchyard switch over to auxiliary power supply. If this action is unsuccessful, systems of essential power supply are disconnected from the grid and switched over to emergency power sources (DGs, batteries).
- Power supply of control, automation and protection systems is provided from batteries. Therefore this function is independent of voltage drop in the grid caused by failures. The entire NPP design applies the principle of electromagnetic compatibility providing functionality of the system in the given electromagnetic environment and in case of disturbance.
- NPP units are operated in accordance with dispatch control of the transmission grid. The transmission grid operator is familiar with conditions and operating limits of NPP, such information is defined in the Transmission Grid Codex. Periodical maintenance of off-site grid components (switchyards, lines, 400/110 kV transformation) and NPP components is coordinated. In case of emergency situations in the power grid (grid decomposition, SBO etc.) recovery of on-site power supply from the transmission grid for NPP is of the highest priority for the grid operator.

1.3.5.2 Power distribution inside the plant

Main cable routings and power distribution switchboards

Supply of home consumption for loads is split among multiple switchyards, power supply systems and sources which have a back-up (on substitution or redundancy principle). This limits consequences of failures of these systems for reactor and unit operation.

Electric loads are grouped according to their importance and in accordance with that they are supplied from sources and grids of corresponding category. Importance of an loads includes the criteria of (safety) function and permissible duration of supply interruption. Functions of loads are classified in accordance with IAAE standard applicable to safety, safety related and safety non essential systems. The following is available for on-site power supply of NPP units:

- **Main sources**, i.e. branch transformers with voltage control (supplied from turbine 1000 MW and/or from 400 kV grid). The operating sources are of purely unit character.
- **Auxiliary sources**, i.e. auxiliary transformers with voltage control (supplied from 110 kV grid). Auxiliary transformers are unit based but can be backed up from the neighboring

unit. The auxiliary transformers are capable to provide power for unit shut down in case of loss of its operating power supply, with preliminary loading by neighboring unit loads.

- **Emergency sources**, which supply essential power supply systems. Emergency sources include diesel generators, batteries and uninterruptable power sources units (rectifiers, inverters). They are installed on site, designed in accordance with requirements of supplied load and their availability is independent of main and auxiliary sources or off-site grid status. Each NPP unit is provided with 3 redundant essential power supply systems classified as safety systems (each of them presents a supporting system for corresponding safety train) and two essential power supply systems for safety related and non safety systems loads supply.

Loads of non essential systems (non safety systems, unit operation systems, power generation systems etc.) are supplied from main power sources. In case of loss of main power supply (branch transformers) automatic switch over to auxiliary sources (auxiliary transformers) is activated. There are 4 main unit 6kV buses in each unit, consisting of two sections. The "a" section supplies RCPs and is provided with the logic of quick controlled switch over to prevent action of reactor protection due to RCP input power drop in case of loss of main power supply. Quick switch over is backed up by conventional switch over due to under voltage. The "b" section supplies secondary components in particular and is provided with conventional switch over due to under voltage.

Appliances essential for nuclear safety (safety systems and safety related systems) are supplied from the essential power supply systems. Essential power supply systems consist of essential power supply grids and emergency sources. They are normally supplied from main or auxiliary power sources. In case of loss of this supply the particular essential power supply system disconnects from the normal power supply grid and switches over to power supply from emergency sources. Uninterrupted power supply of sensitive loads is provided by batteries.

Essential power supply systems disconnection and DG start up is initiated by loss of power supply (U<0,25 Un for 2s, backed up by the logic from off status of breakers in main and auxiliary connection of unit 6 kV bus). Frequency deviations are treated by frequency protection which evaluated frequency drops in 400 kV grid, see the section 1.3.5.1.2. Design analysis and tests proved selectivity of this setting toward automatic switch over function from main and auxiliary source.

These initiation conditions automatically start DGs and they are automatically gradually loaded in accordance with specified ELS program. According to the safety requirements DGs are prepared to be loaded within 10 s from the start command. Function of DGs and automatic system of their loading is verified by periodical tests.

Lay-out, location, and physical protection against internal and external hazards

The unit, branch and auxiliary transformers are situated in front of the turbine hall building. The transformer sites are physically, electrically and fire separated.

Operating sources of home consumption power supply (2 branch transformers, 63/31.5/35.5 MVA each) are supplied from the main generator branch. They supply the main unit 6kV buses, situated in the switchyard building next to the turbine hall. This building contains also step down transformers 6/0.4 kV and 0.4kV switchboards for turbine hall and secondary circuit power supply.

The auxiliary sources of home consumption power supply (2 auxiliary transformers, 63/31.5/35.5 MVA each) are supplied from 110 kV grid. Unit 1 and 2 transformers can provide a back up of each other by 6 kV interconnections and present the source of back-up power supply of 6 kV unit switchyards.

The 6kV unit switchyards supply 6kV motors (e.g. RCPs), 6kV lines of essential power supply and 6/0.4kV transformers for nuclear island loads. These distribution components are installed in the containment enclose building.

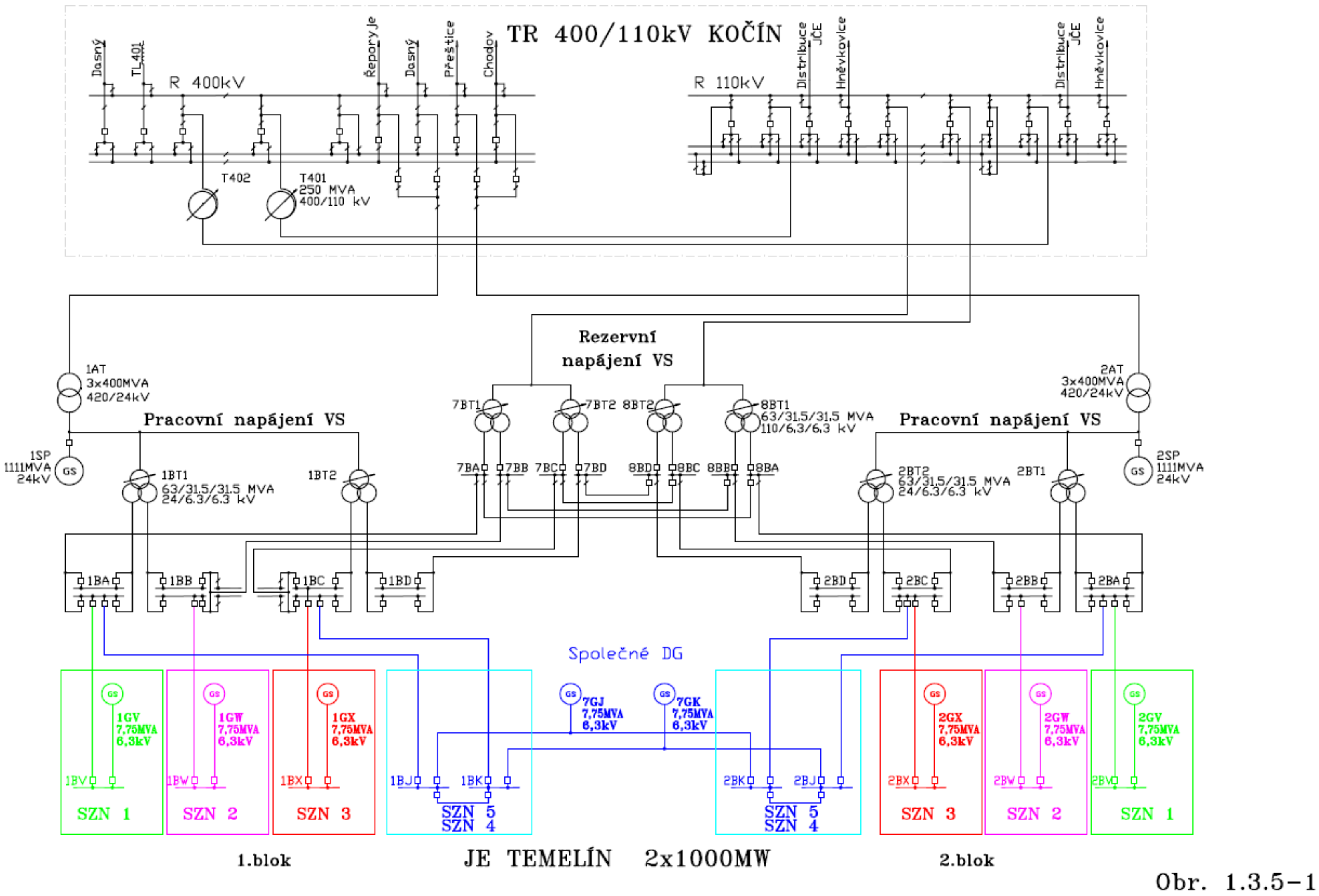
The 6kV unit switchyards also supply 6 kV switchyards, 6kV loads and 0.4kV switchboards in the external buildings (pumping station, compressor station, auxiliary building etc.).

1.3.5.3 Main ordinary on-site source for back-up power supply

On-site sources that serve as first back-up if offsite power is lost

Safety systems of each unit are organized in 3 trains of safety systems (3x100 %). In accordance with this concept each train (identified 1, 2, 3) includes a essential supply system (identified 1, 2, 3) serving as a supporting safety system for power supply for loads of the corresponding division.

Fig. 1.3.5-4 Basic diagram of NPP home consumption power supply



To secure the necessary level of redundancy, these essential power supply systems are independent and separated from each other as far as their layout (structurally as well as from the point of view of fire protection), their electrical arrangement and the control system are concerned. Essential power supply systems 1,2,3 are seismically qualified. Each system includes its own emergency sources (DG, batteries) and power distribution lines. Essential power supply systems 1,2,3 supply also systems of lower safety classification (safety related systems or possibly non safety systems) which require high degree of reliability and redundancy. However these systems must not reduce the safety function fulfillment of safety systems.

Each essential power supply system 1,2,3 consists of the following main components:

- Emergency DG 6.3 kV, 6.3 MW. Diesel generators (GV, GW, GX) have their own storage tanks of diesel fuel designed for operation at full power for minimum 48 hours without diesel fuel make up (in reality this duration is longer since the load is lower). Diesel fuel can be fed from diesel fuel management tanks.
- 6 kV buses of essential power supply.
- 0.4 kV switchboards and 6/0.4 kV step down transformers.
- Rectifiers, batteries, inverters for sensitive loads power supply which require high quality and uninterrupted power supply.

Safety systems trains 1,2,3 and their essential power supply systems 1,2,3 are backed up as a whole (3x100 % redundancy concept). Considering the principle of independence and separation a single failure of one essential power supply system 1,2,3 can not affect availability of the remaining two trains.

DGs are the emergency source of supply for loads with admissible supply interruption for certain period of time (tens of seconds to minutes). DGs start automatically on loss of corresponding essential power supply 6kV bus. At the same time this essential power supply 6 kV bus is disconnected from normal power supply by switch off of two serial sequential breakers. DG loading and operation of essential power supply system and loads are controlled with the highest priority by the automatic system of essential loads sequencer (ELS) in accordance with fixed program without the need of other operator activities. The automatic systems also protect DG from overloading due to incorrect activities of operators.

Loss of voltage is evaluated by two sets. Two signals for loss of voltage evaluation are generated in an identical way in each set. If the value of any delta voltage of three phase system drops under 25 % of the nominal value, any of the signals will act in ELS. If in addition to drop of voltage negative phase component of voltage appears in the system, the signal for ELS start will be delayed for certain period of time. This time presents the time for which a failure of unbalanced short circuit in supply grid type must be switched off by the protections. To improve reliability generation of signals is designed in "negative logic", i.e. ELS input is activated if the signal has the value of logic zero (no voltage).

The generated signal "Category II loss of power supply of 6 kV section" acts for 30 seconds on:

- Activation of latched "Loss of power supply" (LOOP) signal.
- "ELS start" signal - to MCR and ECR (for the duration of latched "Loss of power supply").
- Switch off of both breakers of the section connection to separate power supply of the affected safety system train from normal power supply.
- Switch off of selected loads.
- Generation of blocks of remote manual switch on and automation of normal operation of selected loads.
- DG start (0.2 s delay).

After successful start of DG, achievement of required parameters and successful switch off of at least one section connection breaker the DG automation switches the DG breaker. Switching of DG breaker initiates the loading program (time “zero” of the ELS program). Consequently the selected loads are switched in individual phases of the fixed program. The ELS varies depending on technological conditions of the unit. If RCS temperature is less than 70 °C, then DG loading is controlled by “ELS-cold” program. If RCS temperature is greater than 70 °C, then DG loading is controlled by “ELS-hot” program. The difference between these programs is basically given by the composition of switched loads and the number of stages.

Completion of the “ELS-hot” (after 30 sec) or “ELS-cold” (after 20 sec) programs initiates "End of ELS program" signaling on MCR and ECR. After ELS program completion generation of blocking and normal operation of part of the loads automation is cancelled and for selected loads the blocks of manual switch on remain active depending on DG power output.

Functional tests of DG and essential power supply systems are performed to confirm permanent readiness of safety systems even during unit operation (DG start up test and test of ELS take over of loads after intentional switch off of the sequential switched and simulation of real loss of power supply).

For power supply of the other part of the systems related to nuclear safety and for power supply of the non safety systems, which however provide for general safety of persons and expensive equipment, such as the turbine unit, essential power supply systems 4 and 5 are established in each unit. These supply systems are designed as two subsystems (4.1, 4.2; 5.1, 5.2) which provide back up to each other based on the 100+10% principle.

The main emergency source for these essential power supply systems is the couple of diesel generators (6.3 kV, 6.3 MW each), common to both NPP units. These common DGs and most of the corresponding power supply systems are not of seismic resistance. One operating DG is sufficient for power supply of these systems. This is also the concept of ELS automatics which control gradual loading of these DG. Fuel tank of each common DG is designed for 12 hours of operation at 100% load (supply of all systems in both units). In reality this operating time would be longer in case of simultaneous operation of both common DG at lower load. Each subsystem has its own battery, rectifier and inverter.

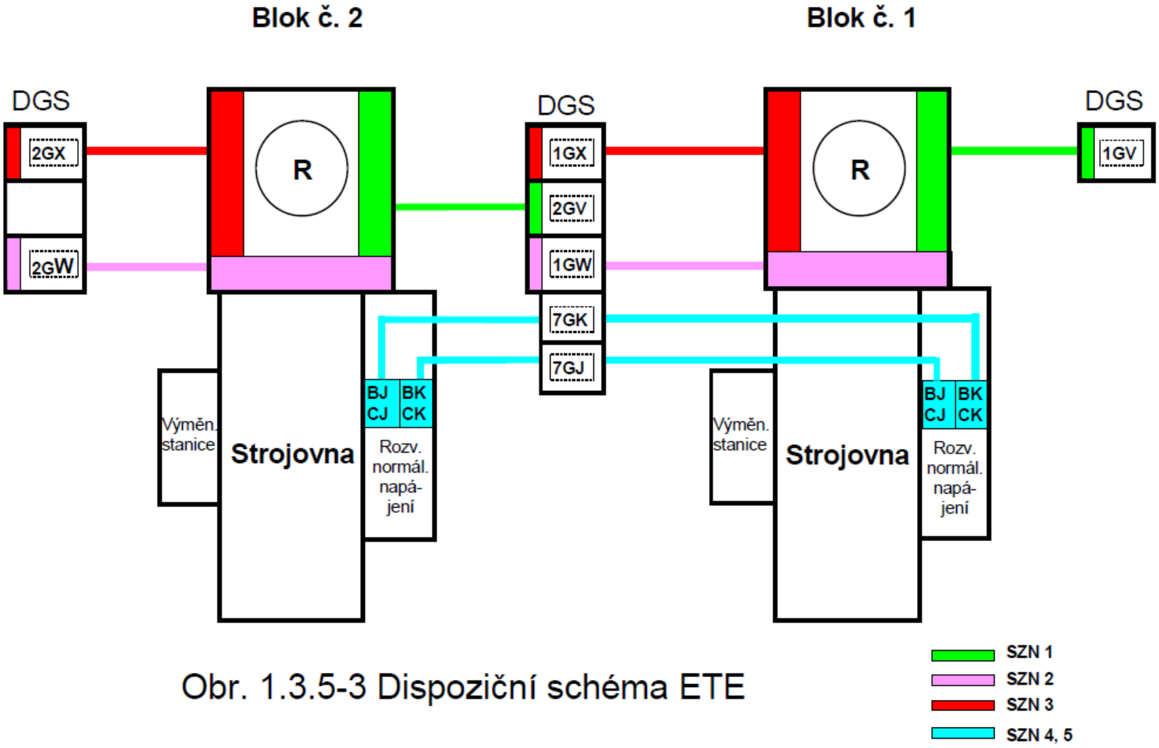
In the SBO procedures these DGs and essential power supply systems are intended as internal alternative power sources. Common DGs are partly diverse (at least due to their layout and connection in the electric diagram as design of control automation) from DG of essential power supply of safety systems. Common DGs can be connected using existing 6 kV lines to switchyards of essential power supply of safety systems 1 and 2.

<u>Redundancy, separation of redundant sources by structures or distance, and their physical protection against internal and external hazards</u>

Essential power supply of safety systems (including DGs) consist of seismic resistance components installed in seismic resistance buildings.

DG buildings for essential power supply of safety systems are of robust steel-concrete structure, protected against simultaneous inoperability due to external risks also by their chessboard layout. DG building for train No. 1 is situated on the other side of the reactor hall than DG buildings for trains No. 2 and 3. DG building for train No. 1 of unit 2 is situated between DG buildings for trains No. 2 and 3 of the unit 1 (buildings of these DG are situated between units 1 and 2). DG buildings for trains No. 2 and 3 of the unit 2 are situated on the opposite side of the reactor hall of the unit 2.

Fig. 1.3.5-5 Layout of power sources and grids



Obr. 1.3.5-3 Dispoziční schéma ETE

Fig. 1.3.5-6 Overview diagram of power supply sources

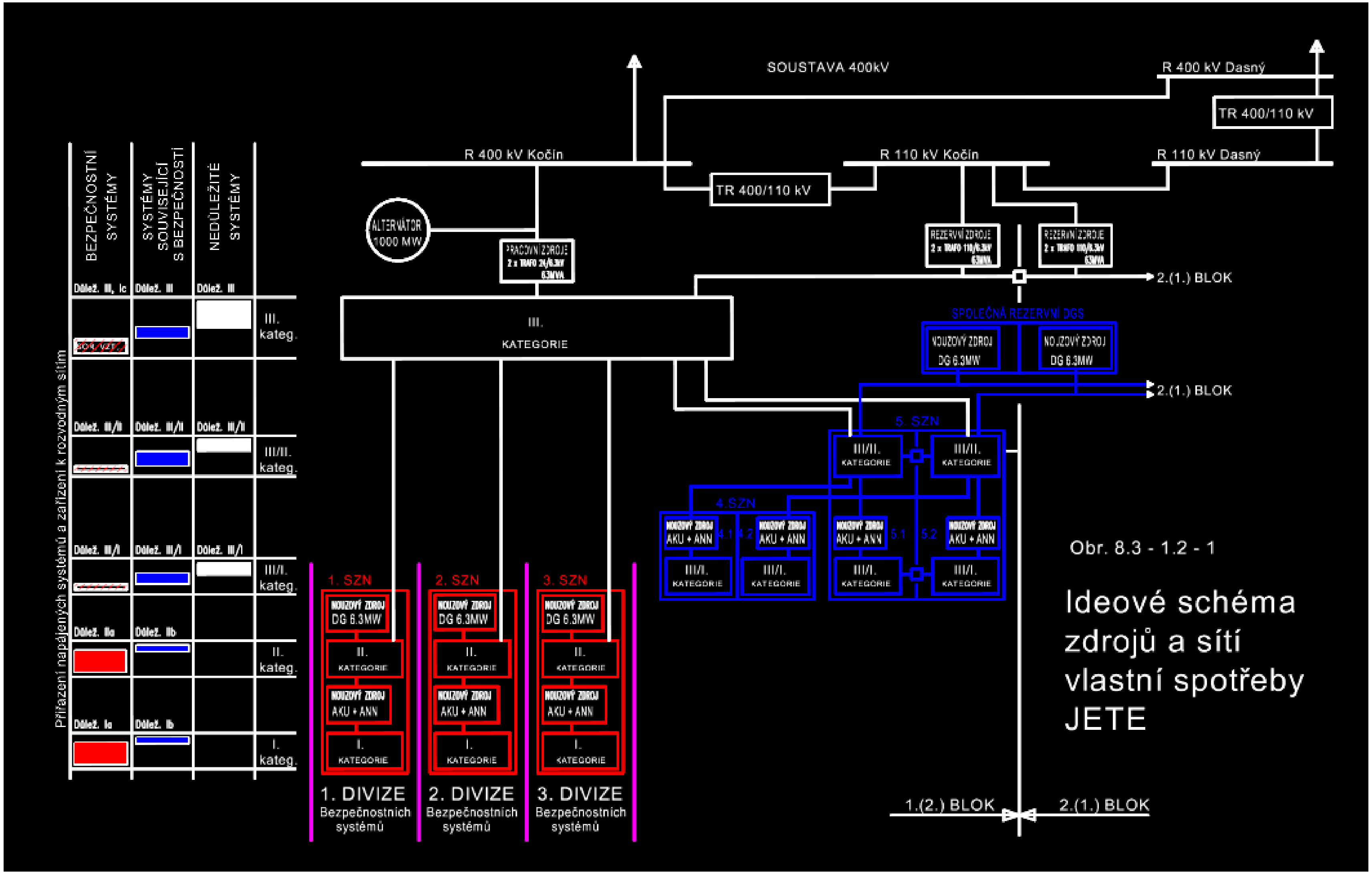
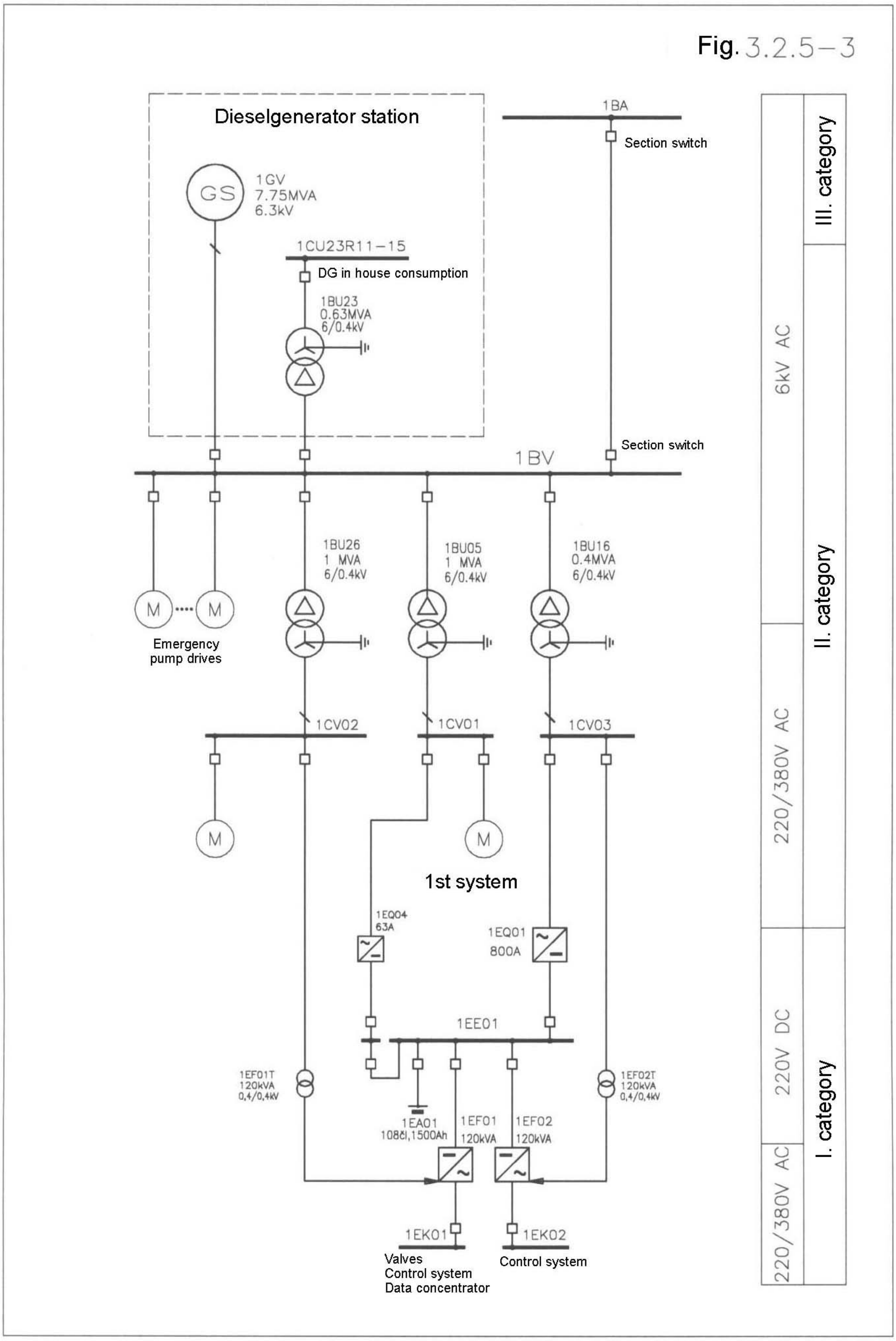


Fig. 1.3.5-7 Home consumption power supply sources



Electric components of each essential power supply system 1,2,3 (6kV switchyard, 0.4kV batteries, ...) are situated on 3 different sides of reactor building to protect them against external hazards.

Cable routes of essential power supply systems 1,2,3 are independent of each other. This provides functional, spatial and fire independence (90 minutes) of these systems and particular divisions of safety systems. Cables in the cable routes are segregated by functional and voltage groups.

All auxiliary systems of the motor and DG (fuel supply to the motor, lubricating oil, internal cooling circuit, boost air, actuation air) are autonomous and, during DG operation, independent of off-site power supply. Each DG has its own distribution lines and power supply sources, including its own batteries. Systems that could be affected by a long-term operation of DG (e.g. clogging of lubricating oil filters) are redundant subsystems whose one section can be shut down during DG operation, conduct the necessary maintenance, and thus prevent a failure of DG due to a loss of auxiliary systems. The quality of diesel fuel is inspected regularly once a month and is maintained in accordance with the applicable requirements.

Essential power supply systems 4 and 5, their common DGs and cable routes are not designed with seismic resistance as a whole. This applies with the exception of essential power supply system 4.1 situated in the reactor building. However these systems also include redundant switchboards and loads are supplied by cables installed in different, separated cable routes. This is due to improvement of reliability and availability in case of fire.

Common DGs are installed in separated rooms of a common DG building in the protected zone between the units 1 and 2. The common DG building provides effective protection against most external effects (adverse weather etc.). Most distribution components of essential power supply systems 4 and 5 are installed in the switchyard buildings (building 500 next to the turbine building).

In addition to tanks situated by DGs (designed for at least 48h of operation for safety DG) there is also diesel fuel storage system available in the NPP, containing up to 2 400 m^3 of fuel oil. If necessary diesel fuel can be supplied to the DG tanks by mobile trucks in case of long term loss of normal power supply.

<u>Time constraints for availability of these sources</u>

Diesel generators have their own fuel oil tanks which are dimensioned, in case of safety DG, for operation at nominal load for at least 48 hours without any fuel oil refill (for a longer period of time in reality). For each common DG at a 100% load (supplying both consumption on both units), the tank is dimensioned to operate for about 12 hours.

With respect to real fuel oil inventory in operating and storage tanks, an operation of safety DG at nominal power is ensured for app. 56 hours. With regard to design of safety systems with redundancy 3x100 % it is possible to gradually use particular safety train and hence to prolong the time of power supply without fuel refilling up to 7 days. All above mentioned times are based on assumption of DG operation at nominal power of 5 MW. The real power of DG (with respect to EOPs actions which require only operation of actual necessary equipment for safe operation) will be app. 2.5 to 3 MW. Such ordinary operating provision can prolong the time of power supply without fuel refilling for additional 40 % up to app. 10 days.

In addition to the tanks installed at the DG, there is also fuel oil storage available on the site with additional fuel inventory 300 m^3 as minimum. Since the fuel oil make-up pumps are supplied from the busses of unsecured power supply, it is necessary to ensure fuel oil refill by mobile means in case of a long-term loss of off-site power. In case of fuel refilling using mobile means it is possible to ensure operation of minimal number of DGs (one safety DG per each unit and one common DG per plant) for additional at least 3 days (with respect to real fuel oil inventory in these tanks of about 1000 m^3 for more than 10 days).

1.3.5.4 Diverse permanently installed on-site sources for back-up power supply

Location, physical protection and time constraints

Since there are no permanently installed on-site power sources designed for case of SBO, analysis of the capability to cope with and recover from SBO has been conducted. The basic methodology of SBO solution was based on US NRC RG 1.155. The purpose of the analysis was to prove that the unit is capable to survive beyond design basis emergency state of SBO lasting for certain period of time, while maintaining safety state.

SBO was solved in accordance with the following definition:

1. The entire NPP (i.e. both 1000 MW units) is affected by loss of main and auxiliary power supply from off-site grid (both 400 kV and 110 kV),
2. In one of the units (identified as "A") the 1000 MW was shut down due not load rejection and DG failure occurred in essential power supply systems in all three trains of safety systems. Batteries remain in operation.

3. Safety of the neighboring unit (identified as "B") is provided by at least one operating train of safety systems.
4. Immediately before or during SBO no design bases accidents or failures occurred. In particular seismic effects, fire or flooding are not considered. All NPP systems, with the exception of systems which caused the loss power supply, work and are available.

For coping with SBO the method of usage of alternating power source is used. The analysis identified the following main limits and requirements for SBO solution:

- Using the methodology in UR NRC RG 1.155 the maximum period for which Temelín NPP should prove resistance against SBO is 8 hours.
- Power supply of the essential service water system and SG emergency feeding must be provided until SG dry out.
- In the period app. between hour 2 and 6 of SBO it is necessary to provide spent fuel pool cooling (depending on the amount of spent fuel inside).

If the said measures are provided the unit is capable to remain in the hot state for app. 56 hours with regard of the water inventory in the SG emergency feed water system.

If it is necessary to bring the unit to the cold state, the following is required:

- Within app. 48 hours after SBO occurrence it is required to put into operation ECCS to increase RCS boric acid concentration and for heat removal from the RCS.
- Containment isolation is required before start of the transition to the cold shutdown state, since during pressure reduction in the RCS and in case of “feed and bleed” method application coolant leakage to the containment will occur. It is required to put into operation all necessary valves of emergency RCS emergency venting system, necessary valves on the lines between the RCS and hydro-accumulators, provide operation of the spray system for pressure reduction in the containment.

For all above measures it is necessary to maintain functionality of required I&C systems and provide functions of particular supporting systems.

- Ventilation for cooling of SG emergency feed water system rooms and ECCS rooms, electric system rooms, and I&C. According to the analysis unavailability of ventilation would result in critical temperature increase in some I&C rooms in more than 60 minutes after SBO occurrence.
- Supply of electric systems (of category I. and II.).

The nuclear island requirements resulted in the need to recover operation of at least one train of safety systems 1,2,3. The following performance balance has been determined by summary of the performance requirements:

- To maintain the unit in the safe hot shutdown state:
  Pp=2,5 MW (max. motor Pn=800 kW) (power supply recovery within 35 minutes).
- To maintain the unit is safe hot shutdown state followed by transition to cold shutdown state:
  Pp=2,5 MW (up to 35 minutes) and consequently additional 1.2 MW (max. motor Pn=800 kW).

Diverse sources that can be used for the same tasks as the main back-up sources

Based on the conclusions of the above mentioned analysis the emergency operating procedure has been developed for Temelín NPP, dealing with SBO type accident with regard to available electric power sources and with regard to actual status of technology. Procedures are available for performance of activities leading to provision of safety systems power supply, considering different possible status of both units' electric diagrams. Therefore the method of solution depends on condition of both Temelin NPP´s units and condition of the off-site grid. The following on-site ACC sources are considered:

1. 6 kV switchyards of the neighboring unit which after loss of power supply from 400 kV and 110 kV grids rejected load on home consumption. The interconnection is assumed via 6 kV back-up power supply lines. The other possible way is interconnection via essential power supply system 5 switchyard.
2. Common DGs (7GJ, 7GK) with the capacity of 6.3 MW. It is obvious that from power output point of view only one of these DGs is sufficient to cope with postulated SBO of one unit. Its power output is probably sufficient even for coping with SBO of both units. The advantage of this source is its stand-by readiness and availability for SBO solution on the site, good protection against adverse weather and full independence on the condition of off-site 400 and 110kV grids.

1.3.5.5 Other power sources that are planned and kept in preparedness

The following off-site ACC sources are considered for SBO management depending on condition of NPP´s units:

1. Off-site 400 kV and 110 kV grids. Off-site 400 kV and 110 kV may be used only if their function and connection to NPP is not failed. The transmission grid operator applies

operating instructions to recover voltage in off-site grids, which specify the highest priority for recovery of power supply for NPP.

2. Lipno hydro power plant (2x60MW), which can be connected by 110 kV lines to the NPP back-up power supply system. This subject to operability of such external lines and 110 kV switchyards, which are located on the supply line. After completion of SBO analysis the Lipno hydro power plant was selected as the main off-site ACC source and this function has been verified by tests. The Lipno power plant (2x60MW) is capable to black start up. The tests verified capability to provide power supply within 30 minutes. The tests included verification of organizational measures for SBO management, functionality of physic protection system, functionality of communication means, roles and procedures of personnel during SBO. Use of Lipno power plant, situated in app. 60 km distance, is subject to operable condition of corresponding switchyards and 110 kV lines.
3. Hněvkovice small hydro power plant - source of small power output (2x2.2 MW to 2x4.8 MW depending on head of water). Power can be brought to NPP through the Kočín 110 kV switchyard using a 110 kV back-up power supply line.

In the NPP other on-site AC power supply sources are available, but they are not designed for supplying the safety systems as part of SBO management and so far their use in SBO management has not been tested.

- DG for supplying turbine lubrication pumps (power of 200 kW).
- DG for data centre (power of 1 MW).

1.3.6 Batteries for DC power supply

Each essential power supply system is provided with sources and distribution lines providing uninterrupted power supply of sensitive loads. The emergency source is in form of lead accumulators 220 V. Following the Forsmark NPP accident all essential power supply system were subject to inspection and technical modification of setting and coordination of protective and monitoring system which currently provide robust resistance against failure and transients in AC power supply grid.

Under normal operating mode the loads are supplied and batteries are charged by rectifiers from sources of normal power supply. In case of loss of main and auxiliary sources power supply of rectifiers is ensured by emergency diesel generators.

The designed rectifiers are capable to recharge accumulators in less than 8 hours.

1.3.6.1 Description of separate battery banks

Essential power supply systems 1,2,3 (classified as safety systems) are provided with systems which consist of thyristor rectifier (220 V, 800 A), batteries (220 V, 1600 Ah) and two transistor inventors (220/380 V AC, 170 kVA). These systems provide power supply for the most important control, monitoring and protection systems and valves within the given safety system train. Another important load is also emergency lighting of areas of the given safety system division (classified as non-safety system).

Essential power supply system 4 includes two subsystems (4.1, 4.2) supplied from essential power supply system 5 and so from common DGs. Each subsystem includes thyristor rectifier (220 V, 1000 A), batteries (220 V, 2000 Ah) and inventor (220/380 V AC, 170 kVA). These systems are classified as safety related systems and supply loads of the control system which are classified as safety related systems or non-safety system. Subsystems 4.1 and 4.2 back up each other (100 %+100 %), the loads have supply lines from both subsystems.

Essential power supply system 5 includes two subsystems (5.1, 5.2) which may be supplied from common DGs. Each subsystem includes 2 thyristor rectifiers (220 V, 800 A), batteries (220 V, 2400 Ah) and inventor (220/380V AC, 170 kVA). These systems are classified as safety related systems and supply loads of the control system which are classified as safety related systems or non-safety system as well as turbine control system. Subsystems 5.1 and 5.2 back up each other (100 %+100 %), most loads have supply lines from both subsystems.

Other batteries systems essential for safety are installed in the diesel generator stations. They consist of rectifiers and 24 V accumulators. They are supplied from corresponding DGs. It supplies control systems and DG protections, the discharge period with this load exceeds 8 hours. They are classified in the same level as DGs (i.e. DG of essential power supply systems 1,2,3 as safety systems, common DGs as safety related systems).

In addition there are more batteries systems in the NPP. Batteries of reactor control rod drives (110 V, 1200 Ah) stabilizes this system in case of short term voltage drops, which may occur in the transmission grid or in the home consumption power supply system. Two 24 V, 600 Ah batteries provide uninterrupted power supply for monitoring of drop of control rods to the lower position.

The conceptual design of NPP in relation to batteries is based on IAAE requirement 50-SG-07: 1982, i.e. discharge period at least 30 minutes. The batteries have been designed and delivered in accordance with this requirement. Within the next development of the design the load of the accumulator system has been reduced by change of the control system to Westinghouse system in particular. It resulted in prolongation of the discharge period but also in significant imbalance of these periods.

Tab. 1.3.6-1 Batteries discharge times in design basis modes

Supply system	**Batteries**	**Battery properties**	**Discharge period [min]** [2) 3)]	
		Lead station battery	LOOP + no failure	LOOP + failure[1)]
1,2,3	EA01,02,03	108 cells Vb2415, 1600 Ah	**> 110**	**N/A**
4.1, 4.2	EA04,05	108 cells Vb2420, 2000 Ah	**500**	**240**
5.1, 5.2	EA51,52	2x108 cells Vb2412, 2400 Ah	**200**	**95**
SOR 24 V	EA21,22	12 cells Vb2312, 600 Ah	**> 600**	**450**
SOR 110 V	EA09	54 cells Vb2412, 1200 Ah	**115**	**N/A**

1) Battery failure of one of the subsystems is considered. Operable batteries supply the full load.

2) The specified discharge period includes supply voltage check of load supplied from the batteries (stabilized and at load peaks) as well batteries check from discharged capacity point of view.

3) The most adverse load time profile is considered for specification of the discharge period. Accumulator capacity is reduced to 72 % C_{10} to respect ageing factor (0.8) and impact of minimum temperature in the accumulator station (0.9).

In addition discharge periods for beyond design basis SBO mode are specified. There are two options - without load disconnection and with load disconnection (i.e. disconnection of part of less essential loads). The batteries discharge period is much longer with controlled relieve of the batteries. Controlled load disconnection is described in the instructions for TPS.

An alternative check has been conducted for safety system batteries, considering the following:

a. Partial batteries unloading (25 % relieve after 30 minutes).

b. No accumulator capacity reduction (100% C_{10}).

Under such conditions the batteries are capable to supply loads for more than **4 hours**.

Tab. 1.3.6-2 Batteries discharge times in beyond design basis SBO mode

Supply system	Batteries	Battery properties	Discharge period [min] [1) 2) 3)]
1,2,3	EA01,02,03	108 cells Vb2415, 1600Ah	**>130 / 260** [4)]
4.1, 4.2	EA04,05	108 cells Vb2420, 2000Ah	**>540**
5.1, 5.2	EA51,52	2x108 cells Vb2412, 2400Ah	**>200**
SOR 24V	EA21,22	12 cells Vb2312, 600Ah	**>600**
SOR 110V	EA09	54 cells Vb2412, 1200Ah	See the design basis modes

1) No failures are considered, all batteries are operable and supply the load.

2) The specified discharge period includes supply voltage check of load supplied from the batteries (stabilized and at load peaks) as well batteries check from discharged capacity point of view.

3) The time profile of loading in SBO mode is considered for determination of the discharge period. Batteries capacity is reduced to 81% C_{10} to respect ageing factor (0.9) and impact of minimum temperature in the accumulator station (0.9).

4) Without disconnection / with disconnection of loads

It results from the analysis that batteries are not critical for management of beyond design basis SBO mode by AAC source connection, because:

a) They are connectible to AAC source which provides supply of loads and charging of accumulator by rectifiers.

b) batteries discharge periods are longer than the period for which it is necessary to recover power supply of loads from AAC source. This applies even at the end of battery life.

1.3.6.2 Consumers served by each battery bank

Direct current power supply is important for function of I&C system (parameter control and monitoring) and for supply of systems necessary for completion of required safety activities, i.e. start of DG and power supply recovery, isolation of RCS letdown paths and other paths of RCS coolant removal, pressure control in SG and RCS and containment isolation.

Among the most important loads supplied from safety system batteries belong:

- I&C systems of safety systems (PRPS, PAMS, PACHMS)

- The steam dump to atmosphere
- Valves of SG emergency feedwater system
- SG safety valves
- RCS letdown paths valves
- Hydro-accumulators isolation valves
- PRZR relief valve isolation valve

1.3.6.3 Physical location and separation of battery banks

Essential power supply 1,2,3 battery systems and related rectifiers and inverters are installed in the reactor building. These are seismic resistance systems, installed in seismic resistance, physically separated rooms.

The battery subsystem 4.1 is installed in the reactor building and is of seismic resistance type. The battery subsystem 4.2 is installed in the switchyard buildings (building 500 next to the turbine building) and therefore is of non seismic resistance type.

The battery systems 5 are installed in the switchyard buildings (building 500 next to the turbine building) and therefore are of non seismic resistance type.

1.3.6.4 Alternative possibilities for recharging each battery bank

If needed the batteries of essential power supply systems 1,2,3 may be charged also by auxiliary rectifier (63 A) connected to the normal power supply grid. This alternative can be used in case of unavailability of the main 800 A rectifier.

The SBO mode assumes batteries charging from available alternative AAC sources.

1.4 Significant differences between units

Both units of the Temelín nuclear power plant are of the same type and there are no differences between them significant from the point of view of safety.

1.5 Scope and main results of Probabilistic Safety Assessments

PSA analyses in JE Temelín, units 1 and 2, were conducted in 1993 -1996. The NPP analysis project covered Level 1 PSA for on-power as well as off-power conditions, including outages, the risks of off-site events, the risks of seismic events and, consequently, also Level 2 PSA.

The original probabilistic models were updated in 2002 - 2003 so as to capture the actual design condition on the commissioning date of the units when all safety improvements were implemented (see chapter 1.2.2). The update also included an analysis of fire risks and flooding risks and an update of Level 2 PSA models. Level 2 PSA analysis currently includes only on-power operation.

Reliability data specific for the NPP site were updated in 2010, replacing the previously used generic data. This resulted in a slight improvement of the total CDF.

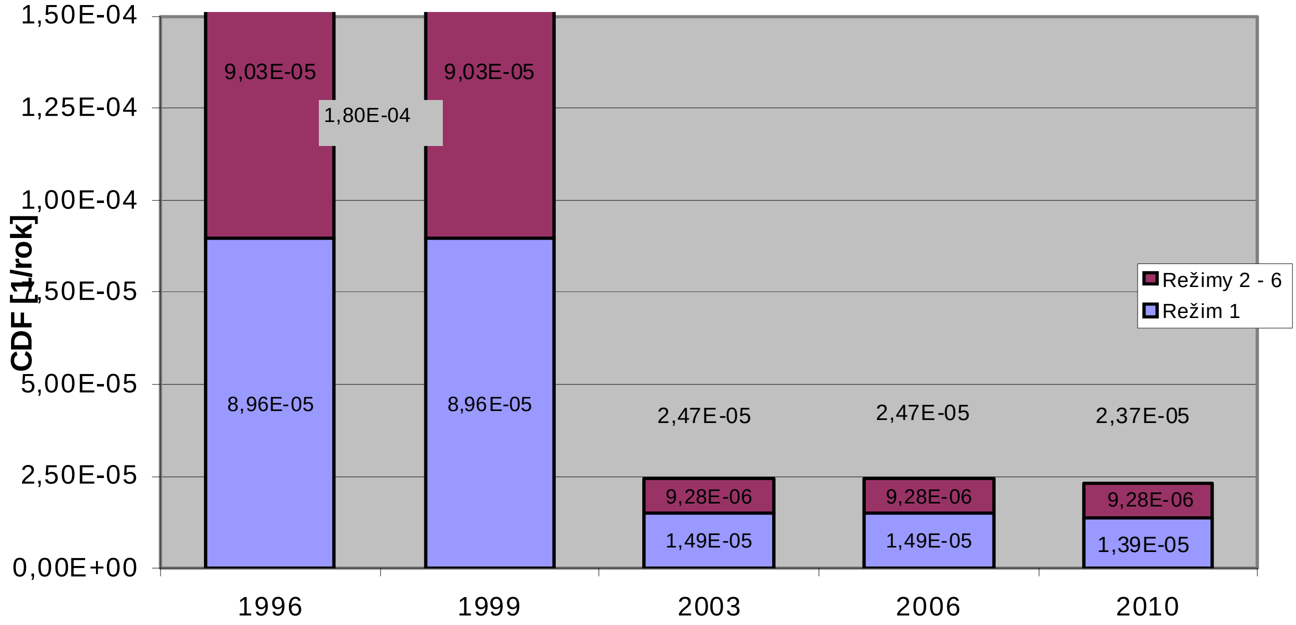
PSA for JE Temelín was subject to the inspection missions IPERS IAEA in 1995 (Level 1 PSA, on-site initiation events) and in 1996 (fires, floods, off-site events and Level 2 PSA). Another IPSART mission took place in 2003, when this analysis was updated. Likewise, an independent evaluation of the specified PSA study, initiated by SÚJB, was conducted by the Austrian firm ENCONET Consulting in 2005; annual inspections and PSA evaluations are performed by SÚJB.

The PSA probabilistic models are updated regularly as part of the Living PSA concept, adopted by the operator, as well as in consequence of SÚJB's requirements on the regular update of PSA models in NPP to make sure that their results reflect the current condition of the JE and meet the basic requirement of their usability for risk-informed applications.

Development of results of Level 1 PSA (CDF) for internal initiation events:

1. At-power operation (Mode 1)
2. Zero-power operation modes/outages (Mode 2 to Mode 6),

Fig. 1.5-1 CDF for internal events



CDF - Average value of Core Damage Frequency for at-power and non-power operation

In the total value of CDF, the category of Large Early Release Frequency (LERF) has a share of 27.11%.

Probabilistic models for the monitoring of the risk level in real time, i.e. Safety Monitor, were prepared in 1996 - 1999, updated in 2003 and put into operation in 2004. They are used to identify and monitor risky configurations of both units during outages and to monitor the risk profiles in real time during operation as well as during outages of both units. They are also used to evaluate operating risks in order to implement risk-informed applications.

Based on the current knowledge, taking into account off-site events in a design-basis scope, the following conclusions from Level 1 PSA apply:

- The share of a seismic event in the CDF risk is below 1E-7 a year
- The share of other off-site initiation events in the risk is negligible (CDF is in the order of 1.0E-7 a year)
- The share of accident sequences resulting in SBO due to on-site causes, i.e. after LOSP, is in the order of 1E-6 a year.

2 Earthquakes

For correct understanding of the following text it is necessary to study content of the chapter 1.3, describing plant systems providing main and supporting safety functions of NPP.

2.1 Design basis

2.1.1 Earthquake against which the plant is designed

2.1.1.1 Characteristics of the design basis earthquake (DBE)

In line with the global practice, there are used two design-basis earthquake levels:

MDE Maximum Design Earthquake, or SL2 Earthquake under IAEA Safety Standards Series No. NS-G-3.3 and NS-G-1.6, which in US terminology corresponds with SSE – Safe Shutdown Earthquake.

DE Design Earthquake, or SL1 Earthquake under IAEA Safety Standards Series No. NS-G-3.3 and NS-G-1.6, which in US terminology corresponds with OBE - Operating Basis Earthquake.

Tab. 2.1.1-1 Binding seismic values

DBE	Level	Acceleration (PGA)	Duration	Comparable $I_{stav.}$
MDE	$SL2_{hor}$	0,1 g	4 - 8 sec	7°MSK-64
	$SL2_{ver}$	0,07 g	4 - 8 sec	
DE	$SL1_{hor}$	0,05 g	4 - 8 sec	6°MSK-64
	$SL1_{ver}$	0,035 g	4 - 8 sec	

PGA - Peak ground acceleration in horizontal and vertical directions

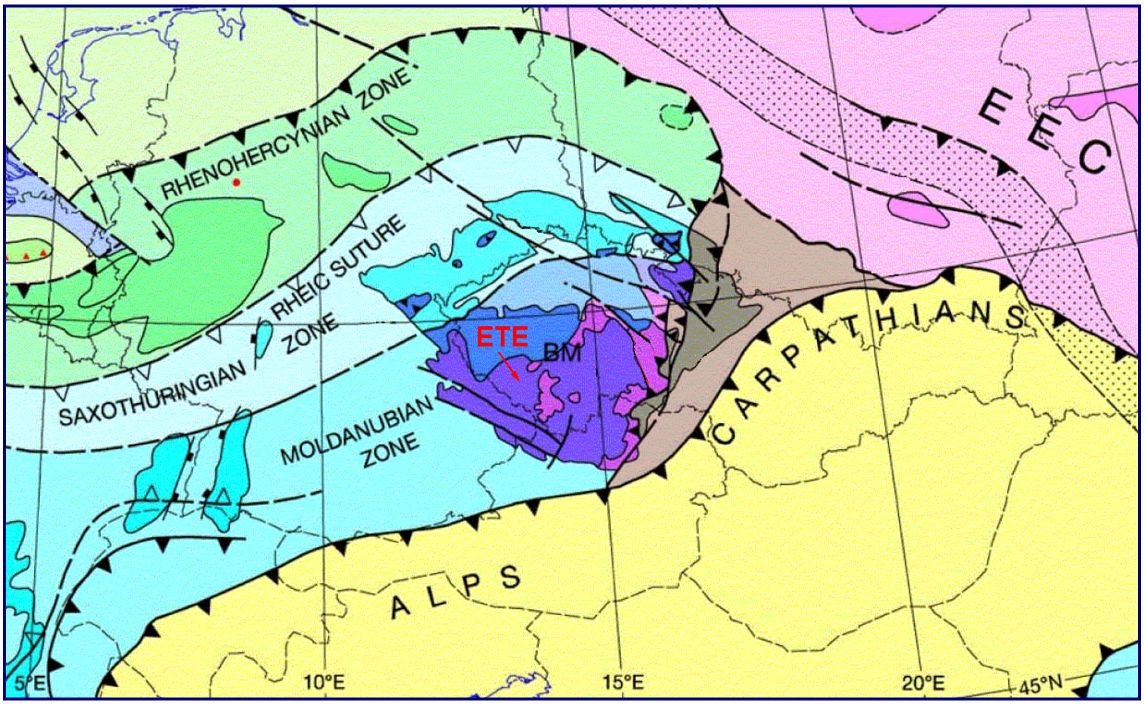
The frequency of occurrence of MDE is projected to be once in 10,000 years, while DE is projected to occur once in 100 years.

Regardless of the acceleration value arising from the site exposure evaluation Temelin design satisfies IAEA requirements (NS-G-3.3, paragraph 2.6) containing minimum value of seismic resistance PGA_{hor} = 0,1 g.

2.1.1.2 Methodology used to evaluate the design basis earthquake

In accordance with IAEA NS-G-3.3 recommendations the Temelin NPP region includes the area within 150 km distance from NPP, while in the south and south-east direction this area was extended up to 230 km. This NPP region extension is justified by generally known low attenuation of east Alpine earthquakes during their proliferation to the Bohemian Massif. The region extends to two basic geological units of the geological structure of Europe - Moldanubica and Alpid. The Temelín NPP location is situated within the Bohemian Massif which forms part of European gercynian orogen. European gercynians present the remaining broad, deformed, mobile zones, impact between East-European or phenosarmat platform, bordered on the west by epicaledonian platform, and between the northern edge of European alpids.

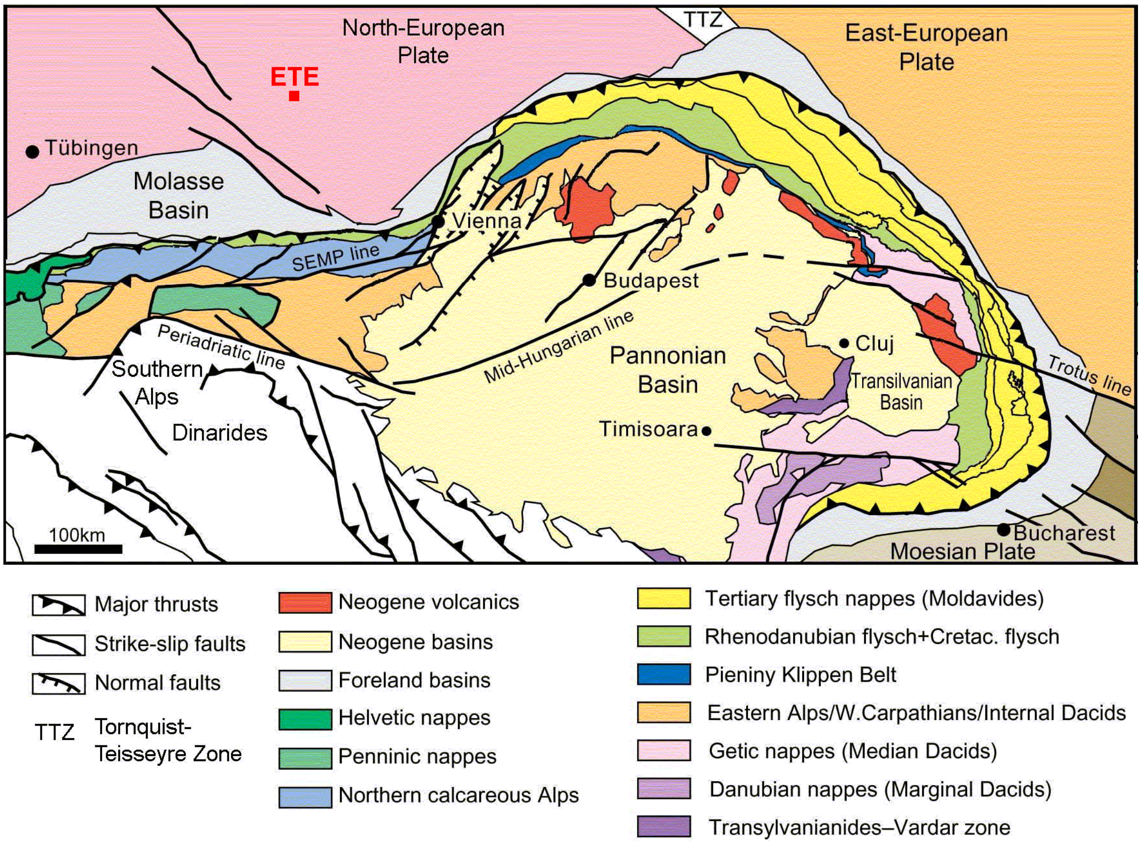
Fig. 2.1.1-2 Basic structures of the Central Europe



Alpids form the Alps to the south of the Bohemian Massif and the Carpathian Mountains on the south-east and east. Deep contact of alpids with their platform forefield is areal. It consists of flat tectonic overlay of nappes and blocks of Alps and Carpathian Mountains to the south edge of the platform. According to geological and geophysical proofs, confirmed by bores, the down-going gercynian platform extends to the distance up to 30-40 km from the front of alpids under Alps. The surface boarder is related to the coverage of deformed formations of Alpic-Carpathian system, identified as the Alpic front or Alpic-Carpathian fore-deep. It extends from Genova, via the arch of Switzerland Alps between Bern and Zurich, along Danube in Austria, to Znojmo, Ostrava, than bends under Krakow and continues to the south along Carpathian Mountains where it ends in the arch of Eastern Carpathian

Mountains by Danube. The Bohemian Massif north border, in face of platform units, is given by number of deep fractures.

Fig. 2.1.1-3 The basic structural units of Eastern Alps and Carpatian Mountains



The first evaluation of the seismic exposure of the site was carried out in 1979. Based on a probabilistic evaluation of the catalogue of historical earthquakes, it was determined that 5.5°MSK-64 would not be exceeded with a probability higher than 90%. Based on a real evaluation of the seismic activity in the region using different methods (seismo-statistical, seismo-tectonic, non-zonal) and surveys conducted from the 1970s, the following earthquakes were determined and confirmed for the original design in 1984:

MDE = 6°MSK-64, with an acceleration value PGA_{hor} = 0.06g

DE = 5°MSK-64, with an acceleration value PGA_{hor} = 0.025g

The design-basis values for an earthquake on the site were reviewed in 1995 in connection with recommendations submitted by an IAEA mission aimed at evaluating the safety of the site. Based on the updated input data and the recommendations contained in IAEA's regulations, the values of MDE and DE were determined in accordance with the values given in 2.1.1.1.

Three different approaches, with a seismo-statistical approach in two different variants, were used to determine the proposed parameters for an MDE (SL2) level earthquake. The resulting values were determined on the basis of a comparison of the results for all of the approaches used as the most conservative values. The use of a combination of these approaches is designed to eliminate any inaccuracies in the earthquake catalogues, generalize the epicenter plans and increase the reliability of the final results.

- Seismo-statistical (probabilistic) - prepared in two guidelines using the same earthquake catalogue, but with a different structure of epicenters.
- Seismo-geological (seismo-tectonic) - based on the assumption that the earthquake epicenters are connected to active faults.
- Experimental - described as a "no-zone method", which does not require any definition of source zones and their boundaries or the determination of seismicity parameters and their seismic potential. It is based on the measurement of actual attenuation characteristics on the epicenter - assessed site route.

<u>Seismic-static approach - method 1</u>

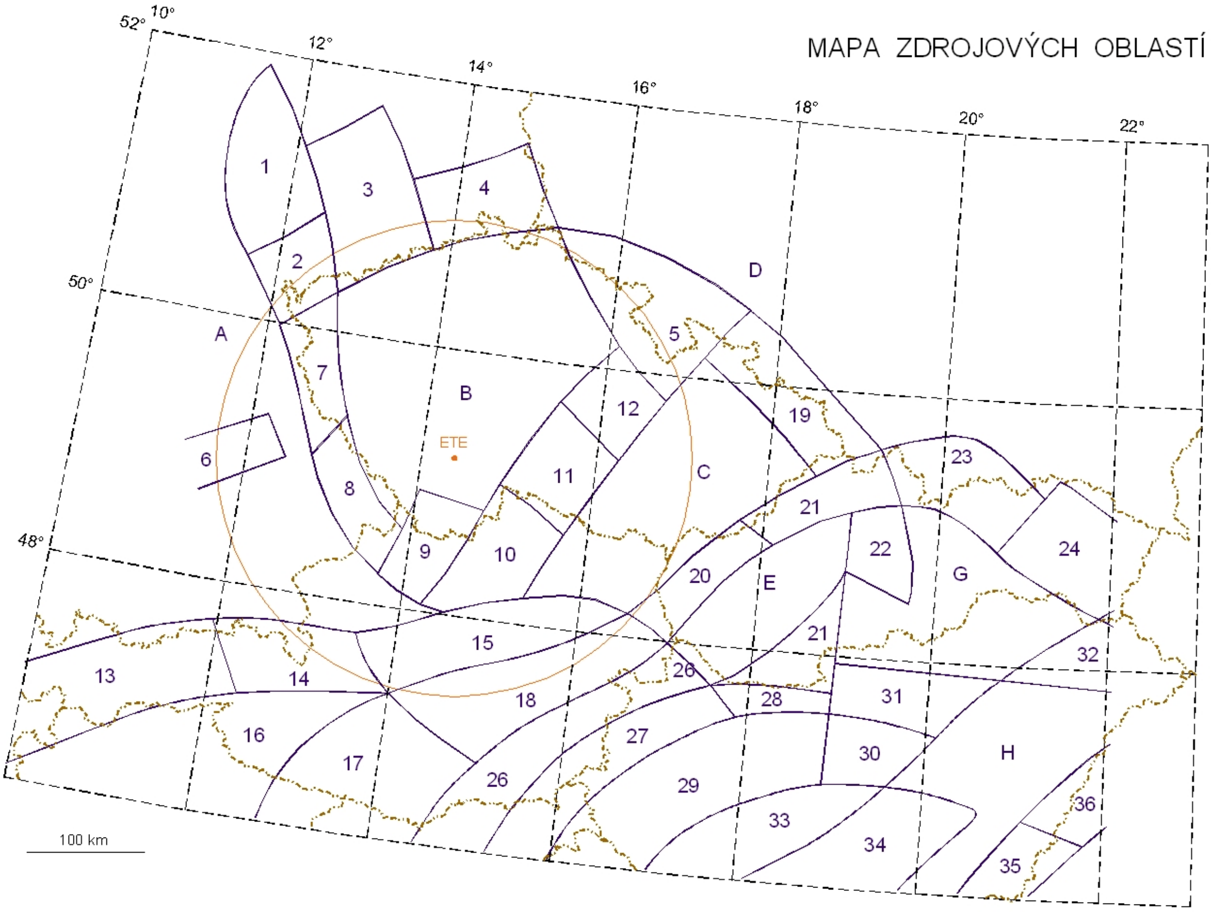
In determining the seismic risk stability of both tectonic and seismic-generating processes is assumed, i.e. it is assumed that so far observed seismic activity trend will remain also in the future. In calculations it is assumed that an earthquake may occur at any point of any area or active section of a fracture up to the maximum possible earthquake for this area or fracture.

From safety point of view the worst case has been considered with regard to the intensity of maximum possible earthquakes in individual seismic focuses, and with regard to the shortest epicentral distance between the boarder of seismic focus areas or an active section of a fracture and the location.

In accordance with IAEA 50-SG-S1 instruction and IAEA NS-G-3.3 recommendations the following two approaches were applied to determine seismic risk of the location:

- An expert estimation based on seismic zoning map.
- Probability estimation based on theoretical mathematical model.

Fig. 2.1.1-4 Map of source areas in the Central Europe



Data of the maximum calculation value of macroseismic intensity for the subject site depending on the focus areas for Temelin NPP site were identified by weighing of focus areas, values of maximum possible earthquakes which may be produced in the area within 10,000 years, and the attenuation curve of macro-seismic intensities, constructed by the azimuth of the focus area - locality, while weighing the shortest distance of the focus area from the site (in accordance with currently used methodology in nuclear power industry it is the most conservative estimation).

The data analysis documented that the highest intensities of earthquake in the Temelín site for the 10,000 years time frame are:

6,5° MSK-64 in case of maximum tremors in focus areas 17 and 18

6° MSK-64 in case of maximum tremors in focus areas 6 and 13

5.5° MSK-64 in case of maximum tremors in focus area 15

Seismic-static approach - method 2

The method 2 approach was based on calculation of seismic risk using probability analysis of seismo-static and partly seismo-tectonic input information (probability curves of seismic risk). This method enables probability evaluation of annual occurrence of various sizes of oscillating motion for many years in advance, but also evaluation of uncertainty related to these values.

Seismic events prognosis is based on the following:

- Distribution of source zones in the location and in the region.
- Seismicity of source zones and maximum possible earthquake which may occur in the zone (seismic potential).
- Drop in the size of seismic movement of earth based on distance from the focus to the site.
- Determination of focus areas and their seismicity.

Assumption on validity of historical seismicity parameters also for earthquakes in the future is based on the idea of repeated rough sliding in existing fractures. However experience shows that new focus areas are present in locations where no historical seismicity is documented. This assumption presents one of the uncertainties in the input data.

The source areas of seismic risk include focus areas of historical earthquakes and lineaments of tectonic fractures or their crossings. In the Central Europe, to which the Temelín NPP region roughly belongs, 60 focus areas were determined. Seismicity of these areas is expressed by frequency graphs and values of their possible maximum earthquake (seismic potential).

Other source zones are fractures in the internal part of the Bohemian Massive, characterized by expert estimations of seismicity parameter values. Evaluations of fractures, together with seismic areas present 71 source areas. Analysis of NPP Temelin seismic risk has been prepared based on evaluation of probability occurrence of earthquake in these source areas.

Seismo-geological (seismo-tectonic) approach

For the purpose of evaluation of seismic activity of fractures in the area of interest the fractures are classified into three classes and 6 categories with regard to the magnitude value (M_{max}), which they are potentially capable to generate. Potential of the fractures is evaluated in a differentiated way for individual structural blocks - regional geological units.

Tab. 2.1.1-5 System of fracture classification and their numeric coding

Class	Description	Category	M_{max}	I_0 [°MSK-64]
A	considerable seismogenous line	I	6.5.	9.5.
		II	6.0-6.4	9.
B	considerable seismo-tecnonic line	III	5.3-5.9	8
		IV	4.7-5.2	7
C	seismo-tecnonic line	V	4.1-4.6	6
		VI	3.6-4.0	5

The data (maximum calculation values of macro-seismic intensity for Temelin NPP site in dependence on seismo-active section of the fraction) were identified by weighing the seismo-active fracture map, values of the greatest possible earthquake intensities, which the seismo-active sections of fracture may produce within 10,000 years time frame, and the attenuation curves of macro-seismic intensities applicable to Temelin NPP location while considering the shortest distance of seismo-active sections of fractures from the location (i.e. in compliance with the current method it is the most conservative estimation).

The data analysis documented that the highest intensity of earthquake in the Temelín site for the 10,000 years time frame is 6.5°MSK-64.

<u>Experimental approach</u>

The experimental determination of seismic risk is based on application of so called "zone free method". This method has many advantages, in particular it does not require the definition of the source zones and delimitation thereof, nor determination of seismicity parameters and their seismic potential.

This method is applicable only recently when instrumental records are available for acceleration of earth seismic oscillations in Temelin NPP area as a result of earthquakes in regional distances. The new method does not need to use only subjective macro-seismic data (catalogues of historical earthquakes and izoseist maps). With the use of authentic instrumentation data it is not required to apply uncertainties of the previous methods, which result from various empiric conversions and expert estimations. For example the relationship between acceleration of seismic oscillations and local macro-seismic intensity is subject to uncertainness in the extend of up to two orders. The new experimental method is considered reliable and prospective by experts.

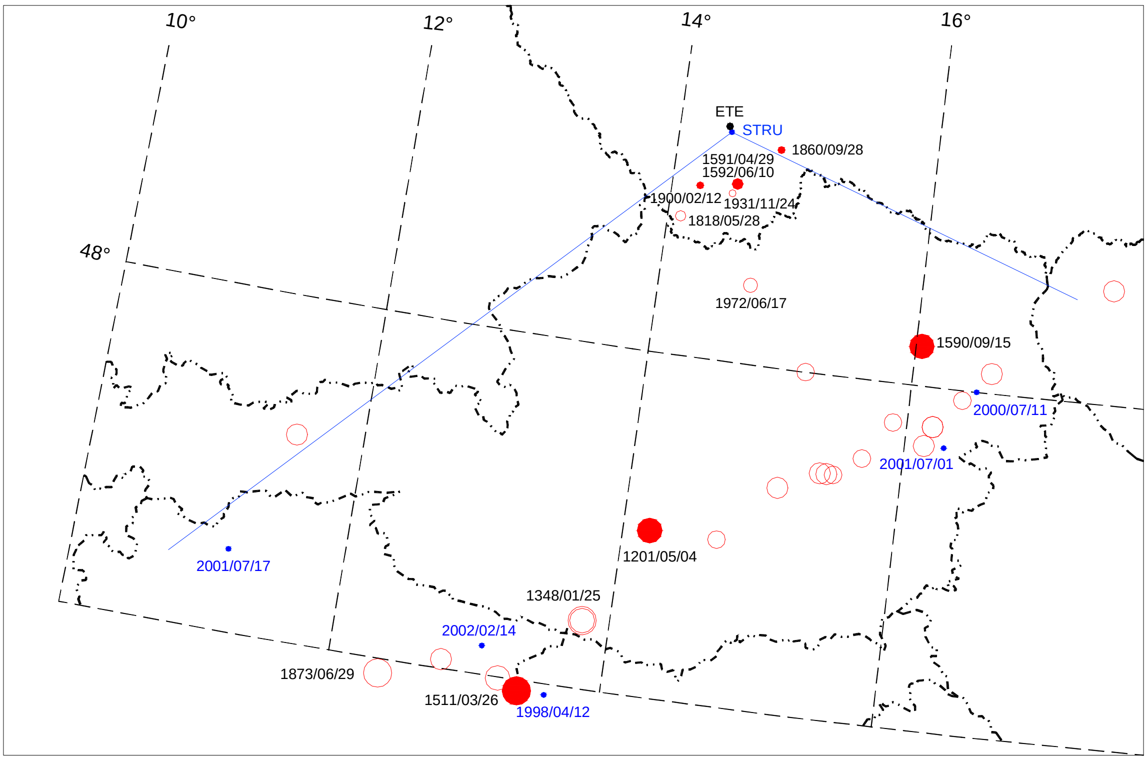
The main sources of Temelin NPPs seismic risk are seismic effects of strong earthquakes in active seismic zones of the region, within the distance above 150 km. For this purpose it is necessary to determine the most appropriate declination relationships for acceleration, applicable for proliferation of seismic waves within the Bohemian Massive. For the first accession such effects may be characterized by the value of peak acceleration in horizontal values, caused by earthquake.

Determination of Temelin NPP seismic risk applying the experimental method is an alternative approach to the above deterministic and probability approaches. The applied experimental method works with the following input data:

- Magnitude catalogue and coordinates of historical earthquakes
- Correction of historical magnitudes and focal distances

Screening of available catalogues resulted in identification of historical earthquakes which after correction application caused acceleration of earth movement in Temelin NPP area at $PGA_{HOR} \geq 10\ cm.s^{-2}$.

Fig. 2.1.1-6 Focal point position of individual earthquakes in Alpine region



Calculations of Temelin NPP seismic risk applying the "zone free method" for the 800 years time frame documented sufficiently conservative determination of seismic risk in application of deterministic and probability approaches.

2.1.1.3 Conclusion on the adequacy of the design basis for the earthquake

The value of peak ground acceleration of PGA_{hor} = 0.1 g a PGA_{ver} = 0.07g for the maximum calculation earthquake and of PGA_{hor} = 0.05 g and PGA_{ver} = 0,035g for the design basis earthquake provides sufficient seismic resistance of Temelin NPP design for this area.

No tectonic structures are found in the territory of the Czech Republic that could produce heavy earthquakes The results obtained from the detailed seismic zoning network underpin the correctness of the seismic evaluation of the NPP site. A continuous evaluation of the epicenters of local microquakes shows in many cases that the causes for these tremors are directly associated with the geological structure of the southern part of the Czech Massif. The earthquake level on the site of Temelin NPP shall not exceed, with a 95 % probability, 6.5°MSK-64 (PGA_{hor} = 0.08 g).

The seismic risk analysis included evaluation of seismic resistance of building and selected systems of Temelin NPP (fragility analysis, seismic resistance evaluation ...). Results of seismic resistance of buildings and selected systems of Temelin NPP documented that the actual resistance of all systems and buildings, essential for safety, significantly exceeds the value of 7°MSK-64 (PGA_{hor} = 0.1 g) determined for MDE (see chapter 2.1.2.2).

The buildings, systems and structures important to perform the safety functions can withstand the level of 7°MSK-64 (PGA_{hor} = 0.1 g) at minimum, thus a safety margin is available for the remaining 5 % uncertainty. The differences in the resistance of specific SSC are individual, yet contribute to a further increase in the safety margin to ensure safety functions.

The Temelin NPP units are provided with a seismic monitoring system. There are action levels set for the individual earthquake levels with respect to the occurrence of an extraordinary event and emergency response actuation. The NPP personnel are well qualified and trained to assess the damage caused to civil structures and components following a seismic event.

Data related to evaluation of location seismic risk are updated on regular basis based on the results of the detailed seismic zoning network

The monitoring results gathered so far can be summarized as follows:

- There has been no earthquake with a magnitude higher than 1 within 40 km of NPP.
- There have been only 9 microquakes with a magnitude of 1 to 2 and no quakes with a higher magnitude within 50 km of NPP.

- By evaluating industrial blasts in local mines, it has been verified that the network is capable of reliably detecting and localizing tremors with a magnitude of 1 to 3 within 50 km of NPP.

2.1.2 Provisions to protect the plant against the design basis earthquake

2.1.2.1 Identification of SSC that are required for achieving safe shutdown state

The building structures and technology needed for performing the fundamental safety functions (reactivity management, heat removal from the reactor core, confinement of radiation and radioactive materials) in case of an earthquake as well as the structures and technology whose damage or failure during an earthquake could result in a secondary exposure of other structures and technology around them, essential for nuclear safety, are classified in category 1 of seismic resistance.

For systems and equipment, seismic resistance category 1 is divided into sub-categories to better define the effects of seismicity:

- Sub-category 1a - Requires the full functional capability to be preserved up to and including the MDE level.
- Sub-category 1b - Requires only preservation of mechanical strength and leaktightness up to and including the MDE level.
- Sub-category 1c - Requires seismic resistance only in terms of possible seismic interactions, in particular maintaining positional stability, up to and including the MDE level. The purpose is to prevent impact on category 1a and 1b systems.

2.1.2.2 Evaluation of SSC robustness in connection with DBE and assessment of potential safety margin

SSC classified within category 1 were subject to seismic analysis, including all safety essential buildings, components, service systems, I&C systems and electric systems, by application of experimental approach, a calculation or an indirect evaluation. They resulted in a proof of their resistance corresponding with the loads based on MDE values.

Tab. 2.1.2-1 Fragility parameters for the selected type equipment

Type equipment	HCLPF
Insulators of outdoor switching stations 400 and 110 kV	0.10g
Fuel storage tank for DGs	0.11g
Switching stations 400 and 110 kV	0.13g
0.4 kV breaker cabinets of safety systems	0.15g
6 kV switchboards / breakers of safety systems	0.15g
6 kV/400 V transformers of safety systems	0.15g
ECW cooling pool	0.17g
0.4 kV cabinets of DG switchboards	0.18g
HVAC fans	0.18g
Clean boron solution tanks	0.19g
SG emergency feedwater tanks	0.19g
ECCS heat exchanger	0.23g
High pressure ECCS tank	0.28g
Emergency boration system tank	0.46g
Containment spray system tanks	0.46g

The calculation of HCLPF (High Confidence on Low Probability Failure) is based on the concept of subsoil acceleration resulting in a "low probability of failure with a high confidence". Subsoil acceleration with the HCLPF value is therefore acceleration for the equipment or building structure at which there is a 95% probability that the probability of its failure will be lower than 5%.

The cabinets of I&C of safety systems are, as a standard, qualified for a minimum acceleration of 0.3g.

The results of the fragility analysis of the structures and selected plant equipment show that the resistance of all safety-related equipment and building structures in which they are installed significantly exceeds 0.1g, specified for MDE.

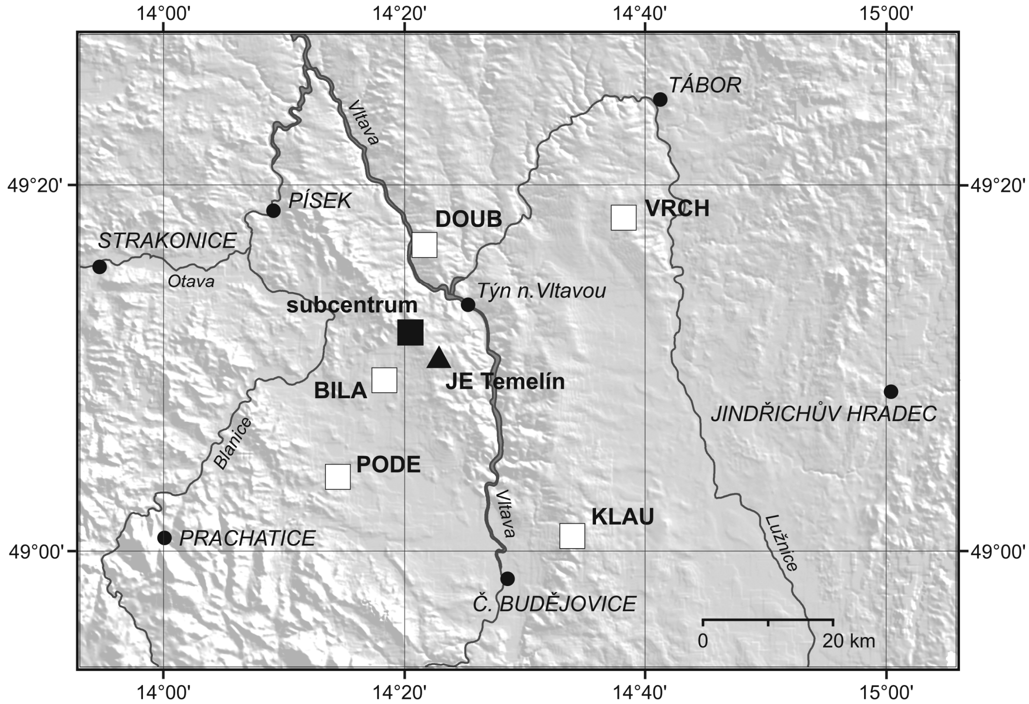
2.1.2.3 Main operating provisions to achieve safe shutdown state

Measurements using the stations of the local seismological network for detailed seismic zoning (DSR) are used to support the results of the evaluation of NPP's seismic exposure.

The main goal of plant detailed seismic zoning network is to register local microquakes with a magnitude of 1 to 3. Seismic phenomena are registered in 4 categories: teleseismic phenomena more than 2,000 km far, regional phenomena 200 to 2,000 km far, close phenomena 50 to 200 km far and local phenomena less than 50 km far. In addition to tectonic earthquakes, the network of stations also registers induced mining tremors and industrial blasts.

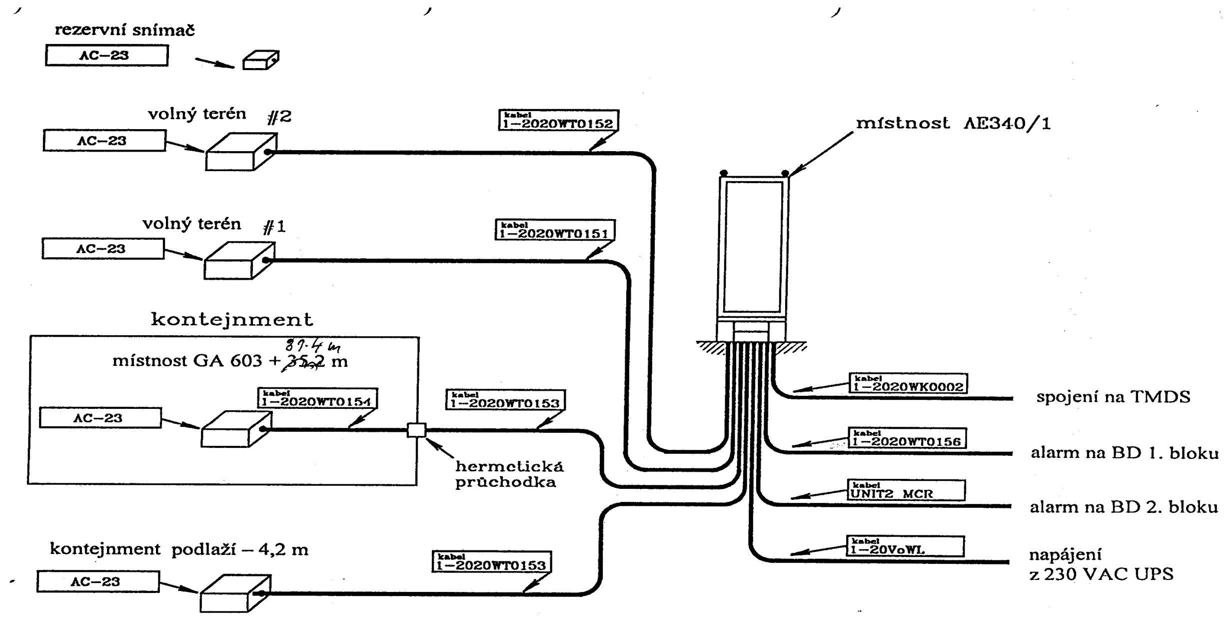
The detailed seismic zoning network around NPP, build and operated under a recommendation from the IAEA, has been under continuous operation since 1 September 1991.

Fig. 2.1.2-2 Map of Temelin NPP DSR network stations



NPP design includes seismic monitoring system (SMS). SMS is always actuated if threshold set value for ground acceleration is exceeded (0.005 g in horizontal and vertical direction for sensors in open ground and basement, 0.015 g in horizontal direction and 0.045 g in vertical direction for sensor inside containment). Along with the corresponding alarms are activated in MCR. No automatic actions of control or protection systems are derived from SMS actuation or seismic alarm. After every seismic event overall plant status evaluation is required. Control plant shutdown is required every time, if MDE value is exceeded or if MDE value is not exceeded but seismic damage are observed.

Fig. 2.1.2-3 Seismic monitoring system



For the individual earthquake levels, the applicable intervention levels are also determined for the purpose of declaration of an extraordinary event and actuation of OHO. Plant personnel is sufficiently qualified and trained to use EOPs and SAMG as well as to perform an evaluation of damage to the equipment after a seismic event.

For all earthquakes that are considered on the site, the performance of fundamental safety functions is not threatened:

a) Reactivity control

b) Heat removal from fuel

c) Confinement of radiation and radioactive materials

A failure to shut down the reactor because of a mechanical defect preventing to a fall of control rods constitutes the so-called ATWS scenario, which has been analyzed for plant even for initiation events that could result from the effects of an earthquake. The reactor would be shut down through the negative reactivity effects, or by the injection of boron into RCS with at least one of the three emergency boron injection systems.

Decay heat removal from the reactor in a hot and semi-hot condition would be carried out in the secondary feed and bleed method by feeding water to SG through the emergency feedwater system and by controlled heat removal to the atmosphere up to the RCS temperature of 150 ^{o}C. There is a sufficient water inventory in the respective tanks for achieving this temperature. After that, heat removal would continue through one of the three redundant residual heat removal systems to the atmosphere through the ECCS exchangers, cooled by ECW, and through ECW cooling pool.

If it was impossible to use the above-mentioned options for heat removal from core, there is the possibility of alternative heat removal from core using the primary feed and bleed method by controlled coolant letdown from RCS to the containment and through the ECCS heat exchangers, cooled with ECW, and through ECW cooling pool to the atmosphere.

Heat removal from SFP would be ensured through one of the three SFP cooling trains through the respective heat exchangers, cooled with ECW, and through ECW cooling pool to the atmosphere.

2.1.2.4 Protection against indirect effects of the earthquake

Side effects of earthquake have been evaluated for NPP site. None of the evaluated effects presented any significant risk for Temelin NPP.

1) Internal flooding may be an indirect effect of an earthquake. Internal flooding was evaluated as part of the analyses of probabilistic evaluation of the flooding risk. For flooding scenarios, where the occurrence of floods could significantly contribute to the risk of core damage and the consequent large releases of radioactive substances, an analysis was conducted with the quantification of their share in the total risk of core damage due to internal flooding. The validity of the assumptions made for these analyses was verified by physical inspection, during which no facts were identified that would be in conflict with or would result in a change in the originally considered assumptions used in the analyses of the risk of internal flooding.
2) The design envisages that a potential severe seismic event could damage seismically non-resistant equipment and structures, which could result in plant disconnecting from the grid and from the supply of media. The design also counts on a loss of off-site power supply 400 kV as well as 110 kV and on the unsuccessful regulation of TG down to home consumption power supply (LOOP). The consequence of this scenario would be the shutdown of both units and heat removal from core would be ensured by natural primary coolant circulation. Power supply to ensure the above-mentioned safety functions is provided by emergency power supply sources (DGs + accumulator batteries), installed in seismically resistant structures. The operating inventory of fuel oil in seismically resistant structures is sufficient for several days of DGs operation. Further fuel oil refill would be provided by tank trucks.
3) In case of a seismic event, the raw water pumping station in Hněvkovice may be lost, as this station, including the supply lines to plant, are not seismically resistant. The

established water inventory on the site, placed in seismically resistant structures (ECW cooling pool), will ensure heat removal from the reactors and spent fuel pools at least for 3x12.5 days.

4) In case of an extensive destruction to the infrastructure and long-term inaccessibility of the site (collapsed buildings, damage to roads, etc.), the alternating personnel might not reach the site during the initial several days. In that case, the required activities would have to be secured by the personnel present on site during the event. Personnel rotation would be settled in cooperation with public authorities (IZS, army, etc.).

5) Access to the essential civil structures could be limited due to destruction of structures not having sufficient seismic resistance onto the on-site access roads, as well as due to falling debris into the area of plant entrance. Access could be also prevented to the emergency control centre and the emergency preparedness shelters. In such a case, it would be possible to use the back-up access road/entrance to the plant and to activate the emergency control centre in České Budějovice. Inaccessibility of shelters, if any, would be settled case-by-case by evacuating the non-essential personnel outside the site. Inaccessibility of the technical support centre could be settled by operating the centre from the main or emergency control room.

6) For the timely indication of on-site flooding as an indirect effect of an earthquake, a water detection system is installed in the rooms and if a water level is indicated in a room, the applicable activities are carried out.

2.1.3 Compliance of the plant with its current licensing basis

2.1.3.1 Licensee's processes to ensure that SSC needed for achieving safe shutdown after earthquake remain in operable condition

The current condition of the plant is in accordance with design requirements. To maintain permanent compliance of the current equipment condition with the design-basis requirements, many regular activities are performed. These activities include:

- Maintaining a seismic qualification of equipment and buildings.
- Round inspection activities to ensure the required equipment condition and avoid equipment damage, accidents and fires or injuries to people and to ensure equipment operation at a high safety level.
- In-service inspections and tests of the equipment.
- Predictive and corrective maintenance of the equipment.

2.1.3.2 Licensee's processes to ensure that mobile equipment and supplies are in continuous preparedness to be used

On the site, there is the Internal Brigade of the Fire Rescue Service consisting of app. 16 firemen per shift (non stop 24 hours service). The Internal Fire Brigade which is equipped with the relevant fire-suppression equipment and which is trained to perform an intervention in any place on the site. fire-suppression equipment and intervention personnel are stationed in a building without any seismic qualifications. To prevent any danger the technical means and the fire brigade personnel would be relocated to open areas in case of tremors or other indirect symptoms signaling.

For pumping and transport of water the fire brigade has available 4 tank-truck fire engines, 1 combined fire truck and 3 trailer fire-engines, with the total nominal capacity of 280 l/s.

2.1.3.3 Potential deviations from licensing basis and actions to address those deviations

Based on special inspections regarding seismic resistance, carried out after an event in the Fukushima NPP in May 2011, no serious noncompliance were identified between the current condition and the design-basis requirements.

2.2 Evaluation of safety margins

2.2.1 Range of earthquake leading to severe fuel damage

On the site an earthquake stronger than 6.5°MSK-64 (PGA_{hor} = 0.08 g) cannot virtually occur. SSC essential for carrying out safety functions are resistant up to the value of 7°MSK-64 (PGA_{hor} = 0.1 g).

Hypothetical earthquakes with effects larger than MDE (7°MSK-64), i.e. exceeding the value of PGA_{hor} = 0.1 g, do not have the nature of cliff-edge effect, taking into account individual safety margins and inherent redundancies of the VVER1000 design. The performed fragility analyses of the structures and selected equipment (see 2.1.2.2) show that the resistance of all safety-related equipment and building structures in which they are installed significantly exceeds 7°MSK-64.

The differences in the resistance of specific SSC are individual, yet contribute to a further increase in the safety margin to ensure safety functions. A fulfillment of safety functions (cliff edge) could be violated at acceleration values of PGA_{hor} = 0.15 g and higher, which are not realistic on the site. Subject to approximation of NPP site seismic risk curves by their prolongation to the higher intensity values the frequency of occurrence of a seismic event of $PGA_{hor} = 0.15g$ and higher intensity can be estimated at less than 1E-8/year. This corresponds to occurrence once per 100 million of years of even less.

In case of an earthquake, SFP integrity would not be breached (these are pools placed in a seismically resistant CNTN, embedded in reinforced concrete with stainless steel lining (pools bottoms are at spot height +20.7 m, upper edges at +36.9 m). In case of a loss of SFP cooling or leaks, there is an alternative way to ensure their make-up and cooling through a route dedicated for this purpose from the discharge of containment spraying system pumps, release to the containment and then through ECCS heat exchangers, cooled with ECW, and through ECW cooling pool to the atmosphere. For detail description of SFP cooling system see the chapter 1.3.3.

2.2.2 Range of earthquake leading to loss of containment integrity

Barriers against radioactivity leakage - fuel cladding, primary circuit pressure boundary and containment - are seismically resistant. The design-basis seismic resistance of the containment structure (pre-stressed reinforced concrete) exceeds the value of design basis seismic resistance with certainty. Isolation of its pipes and locks is provided by redundant isolating components, whose resistance is also at least 0.1 g (with a sufficient margin). The resistance of I&C systems for the automatic CNTN isolation is up to 0.3 g. Based on the designer's engineering judgment, it can be estimated that units are resistant up to the value of 0.2 g (and more). Under all circumstances this resistance is significantly higher than possible earthquakes which may occur in this area.

2.2.3 Earthquake exceeding DBE and consequent flooding

Therefore under no circumstances the site is seriously endangered by any consequent flooding. For details concerning NPP site flooding risks please see the chapter 3.2.1.

The main plant structures, where equipment necessary for nuclear safety is located, are at a spot height of 507.30 m above sea level, which is 135 m above the level of the Hněvkovice water reservoir, which is a part of the Vltava Cascade responsible for regulating the flow rate through the whole cascade.

Safety evaluation was also performed for plant in terms of the potential destruction of the water reservoirs on the upper Vltava River (including an earthquake). There are only two large reservoirs, namely Lipno I on the Vltava River and Římov on the Malše River, and the smaller reservoir Lipno II, upstream of the Hněvkovice reservoir. In case of damage to the

Lipno I reservoir, the flow rate would be at the level of 10,000-year's water in the Hněvkovice profile. Although this condition leads to loss of Hněvkovice raw water pumping station it is documented that the stock of water in the site is sufficient for heat removal from the reactors and spent fuel pools for at least 3x12,5 days.

2.2.4 Measures which can be envisaged to increase robustness of the plant against earthquakes

It is obvious from the previous evaluation that NPP site is extremely well selected from seismic risk point of view. The NPP site can be evaluated as highly stable in relation to external natural events, including seismicity. In addition the robust VVER1000 design and diversity of seismic resistance SSCs provide sufficient resistance and margins against any design-basis and beyond design-basis seismic events.

Any possible adverse consequences of earthquake are therefore limited only to non-seismic resistant SSCs which may contribute to fulfillment of supporting safety functions. This concerns for example the long term power supply after loss of off-site power (3 days and longer) only from emergency sources, associated with necessary external supply of diesel fuel for DG operation.

Activities after a seismic event may be complicated by loss of communication between the control centers and the intervening personnel, including communication with off-site control centers and state authorities, as a consequence of damaged infrastructure around NPP.

The objective of the proposed measures is additional improvement of the defense-in-depth during earthquake.

Areas for defense in-depth improvement are included in the following table. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation. For detail description of the corrective measures please see the Attachment.

Areas for improvement	**Corrective Action**	**Term** (short-term I / middle-term II)	**Note**
Technical means	Alternative diesel fuel supply by truck tanks for long term operation of DGs	I	
Procedures/guidelines	EDMG instructions for use of alternative means	II	

Areas for improvement	**Corrective Action**	**Term** (short-term I / middle-term II)	**Note**
Emergency preparedness	Capability of emergency organization to operate out of Emergency Control Center	I	
Analysis	Seismic resistance of Fire brigade building	I	
Communications	Alternative communication means after seismic event	I	
Personnel	Analysis of risks for shelters during seismic events	II	
Personnel	Personnel security after seismic event	I	
Analysis, technology	Access to buildings, availability of heavy duty technology	II	

3 Flooding

For correct understanding of the following text it is necessary to study content of the chapter 1.3, describing plant systems providing main and supporting safety functions of NPP.

3.1 Design basis

3.1.1 Flooding against which the plant is designed

3.1.1.1 Characteristics of the design basis flood (DBF)

The site is located in a place with a normal occurrence of rainfalls and there are no extremely high, excessive rainfalls there. In terms of drainage, the plant has a cascading arrangement, where structures important for nuclear safety are located at the highest spot height of 507.10 m above sea level, with a descending tendency toward the edge of the site, which provides for a natural, gravity-induced drainage from the site even if the rain water sewer systems fail.

The design-basis values for maximum rainfalls with a repetition interval of 100 and 10,000 years are determined on condition that surface drainage is ensured in the area of the power plant, but the sewerage system is fully out of operation since its gullies are clogged. Considering the realistic hydrological characteristics in the area with the final terrain level at 507.10 m above sea level, the maximum daily total precipitation is essential for assessing its protection against flooding, where the maximum height of the water surface in the area would reach 47.2 mm in case of a 100-year's rain and 88.1 mm during a 10,000-year's rain. In the area with a final terrain level at 504.10 m above sea level, extreme rainfalls will be drained over an edge with the maximum height of 114 mm.

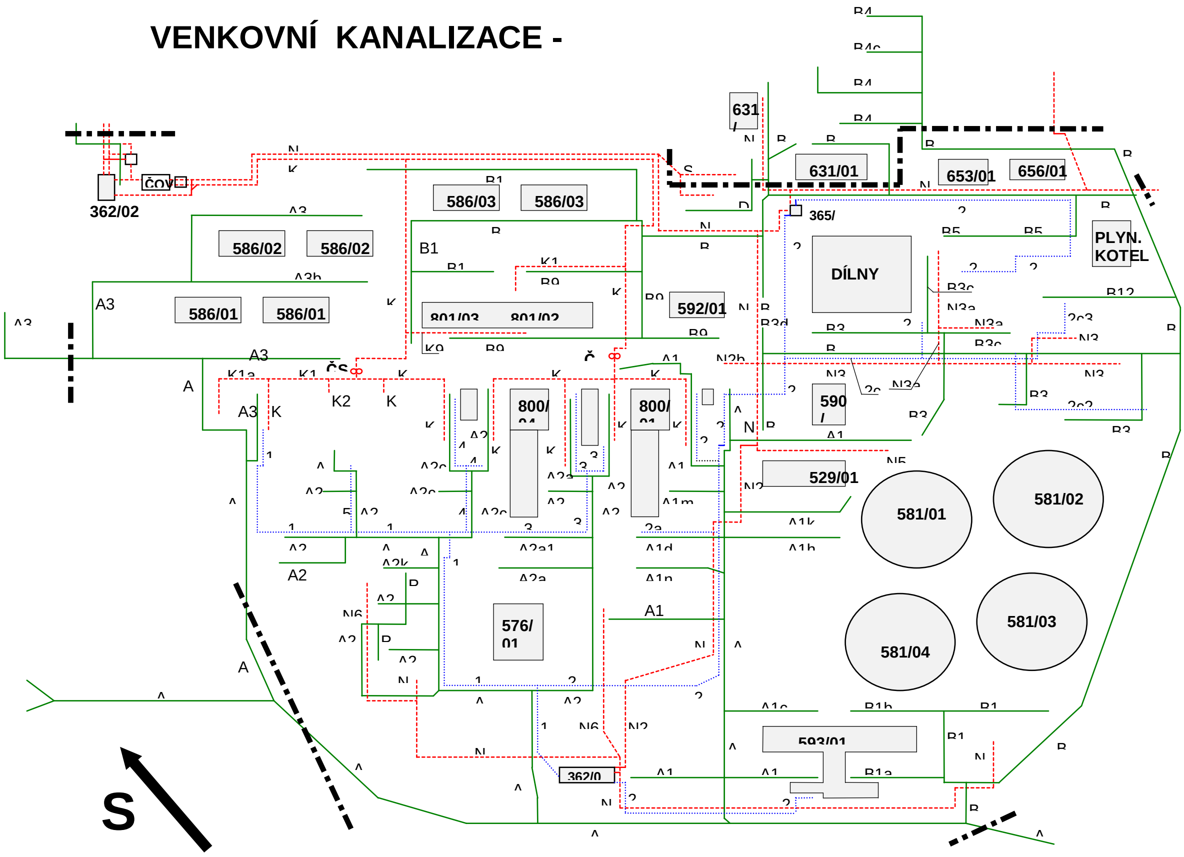
The plant main structures, where equipment necessary for nuclear safety is located, are 135 m above the level of the Hněvkovice water reservoir, which is a part of the Vltava Cascade responsible for regulating the flow rate through the whole cascade. The Hněvkovice dam has been built as part of the service water supply system for Temelin NPP. Therefore it is practically impossible that the site would be flooded as a consequence of increased water flow in Vltava river.

3.1.1.2 Methodology used to evaluate the design basis flood

An evaluation of floods generated on upper streams shows that in the Hněvkovice profile, the water level in case of 10,000-year's water will be about 5 m above the maximum level, which will result in the flooding of a major part of the NPP raw water pumping station. The dam of the Hněvkovice reservoir can then be destroyed. Both these conditions would prevent standard raw water supply to plant and it would be necessary to shut down both units. However, there is a sufficient inventory of water on the NPP to ensure units cool-down to a cold shutdown condition. The inventory is stored in a raw water reservoir, in the tower cooling system and, last but not least, it is possible to feed the essential cooling water circuit from the drinking water supply lines.

When evaluating large flow rates in the Hněvkovice reservoir profile, the condition achieved during the 2002 floods was verified. The maximum water level reached on 13 August 2002 was at 371.56 m above sea level, i.e. it reached a height approximately corresponding with the projected maximum spot height for this reservoir (371.60 m above sea level). Water passage through the Hněvkovice dam was standard and no major damage was identified on the reservoir as well as on the plant pumping station.

Fig. 3.1.1-1 NPP rain water drainage system (identified in green on the picture)



For the design of the rain water sewer network (for removing rain water) in NPP, the selected intensity was 15 minutes of non-reduced rain with an intensity of 127 l.s-1.ha-1. The plant has a drainage area of 133.14 hectares, which is divided between two receivers (receiver A drains the western and southern part, with an area of 80.06 hectares, while receiver B drains the northern and eastern part, with an area of 53.08 hectares). The average drainage coefficient is 0.415 and the total drained volume at the specified intensity of non-reduced rain is 7,020 l.s-1 for 15 minutes.

The Hněvkovice concrete gravitation dam forms a water reservoir with the total capacity of 22.2 mil.m^3 at max. level of 370.50 meters above sea level. All essential structures of the dam are designed for the maximum level at the final elevation of 372.0 m above sea level. At permanent retention with the level at the elevation of 365.0 m above sea level the capacity of the water reservoir is 12.8 mil. m^3. While the water reservoir bottom under the dam body is maintained at 354.0 m above sea level, the water depth at water pumping point for Temelin NPP is 11 to 16.5 m. The main purpose of the reservoir is to create a space for sudden increase flow from Lipno water reservoir, situated 120 km upstream.

Fig. 3.1.1-2 Hydraulic structures in power plant's region

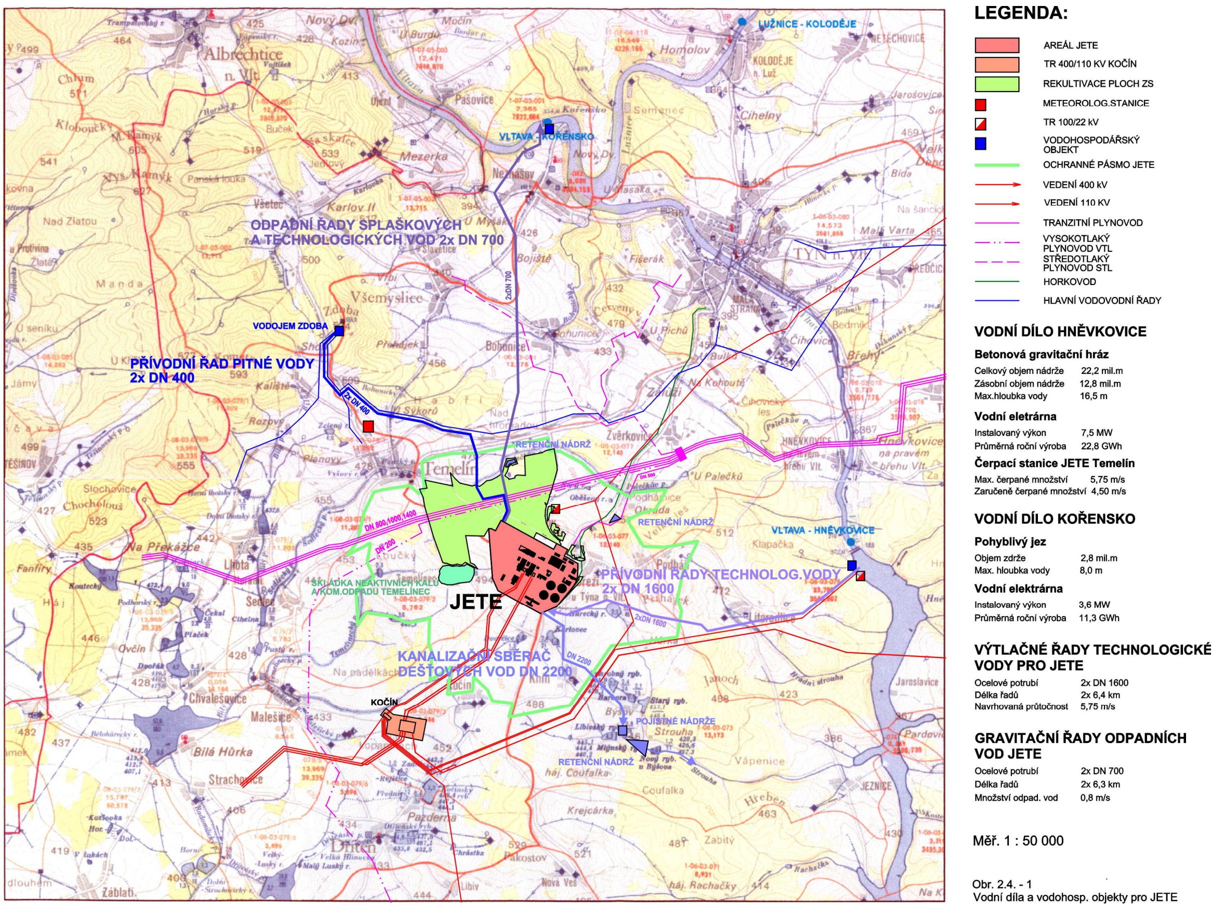


Fig. 3.1.1-3 Hněvkovice dam with raw water pumping station for Temelin NPP

Level drop in Hněvkovice dam below so called dispatching level is a trigger for initiation of increased drainage from Lipno dam to maintain required water supply for plant under all operating modes. Water inventory of Lipno dam is 252 mil. m^3.

There are two large reservoirs, namely Lipno I on the Vltava River and Římov on the Malše River, and the smaller reservoir Lipno II, upstream of the Hněvkovice reservoir. In case the Lipno I dam would be damaged the Hněvkovice profile would be subjected to 10,000 years water flow rate which may cause damage of Hněvkovice dam. Upon culmination of the flood wave, the water surface in the reservoir at the Hněvkovice dam would reach 376.7m above sea level.

Other important function of the Hněvkovice dam is related to sediment and winter mode, when with regard to the water depth in the reservoir, it creates conditions for safe water supply for Temelin NPP purposes under any operating and climatic conditions. Additional purpose of Hvěvkovice water reservoir is also use of power generating capacity of its hydraulic power plant operated in semi-peak mode with daily balancing of natural flow rates after deduction of consumption of the pumping station of Temelin NPP. The hydraulic power plant is considered as one of alternative off-site sources of power supply of Temelin NPP´s on-site power in case of SBO.

The main purpose of the Kořensko weir level is to maintain water level within the end section Orlík water reservoir at the elevation of 353.0 m above sea level, i.e. close to the maximum level of Orlík water reservoir, regardless to water level subsidence in this reservoir. With the water level at normal backwater level, i.e. 353.0 m above sea level, the weir backwater area contains 2.8 mil.m^3 of water.

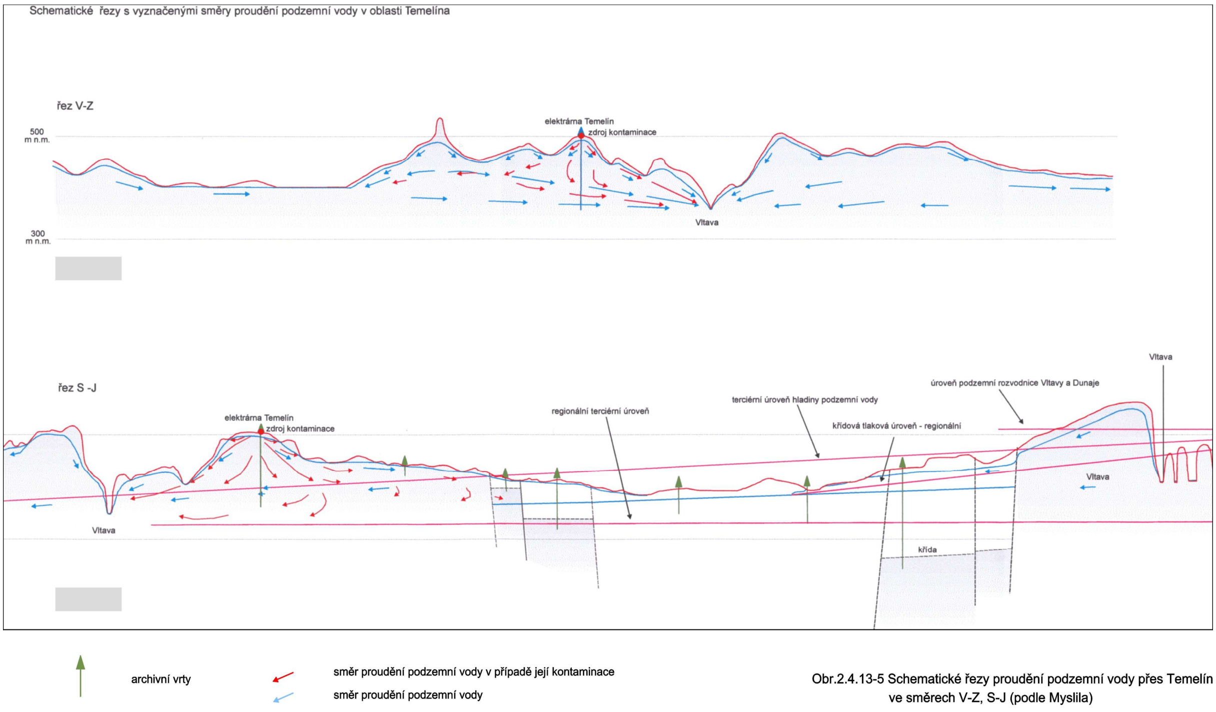
Similarly to Hněvkovice water reservoir hydro power of Kořensko water reservoir is utilized in little hydro plant in tandem with Hněvkovice water reservoir. One of the main purposes of this dam is to create conditions for safe homogenization (dilution) of discharged waste water from Temelin NPP. Waste water, after flowing through little hydro plant in the dumping structure of the waste line, is transferred to Kořensko water reservoir and discharged to draught tubes of this water reservoir. When the power plant is out of operation waste water is discharged to the discharge area of this water reservoir's weir dam section.

Fig. 3.1.1-4 Kořensko water structure

As regards the smaller streams of Rachačka, Strouha, Hradní strouha, Palečkův potok, Bohunický potok, Karlovka and Temelínecký potok, with their sources in the immediate vicinity of plant, long-term observations did not determine any conditions that could result in plant flooding. This assertion is supported by the morphology of the landscape and in particular by a significant gradient of these streams away from the power plant, as well as by hydrological data.

The underground water level on the site is at a depth of about 10 to 12 m below the terrain, i.e. approximately at the level of 500.0 m above sea level. Because plant is located on a plateau and underground water is only fed through rainfalls, underground water is drained from the plant away to all sides. There is no exposure of structures or rooms with equipment essential for nuclear safety due to a shallow underground water horizon.

Fig. 3.1.1-5 Flow of underground water in Temelin NPP location



3.1.1.3 Conclusion on the adequacy of protection against external flooding

Building structures protective measures against the effects of flooding due to the water streams in the vicinity of plant are not necessary considering the prior evaluations and the history of floods in the region. The NPP is not exposed to floods due to water courses.

Only external flooding caused by extreme storms can be expected at NPP site. All SSCs installed in above ground buildings above 507.10 m above sea level, are not endangered by external flooding (with the exception of possible flooding of diesel fuel supply system for the diesel generator station) with regard to design measures (sills, sealing of entry and maintenance openings), which prevent inflow of accumulated water to these buildings. All buildings essential for safety are resistant to flooding up to maximum accumulation level of 88.1 mm, which corresponds to one day extreme storm with the expected repeatability once in 10,000 years. Even in case of theoretically possibly shorter rainfalls with a higher intensity, the whole passive system of rain water gravity sewers is capable of removing the precipitation, taking into account the large volume of the drains and the short duration of such intensive rainfalls.

3.1.2 Provisions to protect the plant against the design basis flood

3.1.2.1 Identification of SSC that are required for achieving safe shutdown state

SSCs necessary for safe reactor shutdown are described in the chapter 3. Considering measures which provide prevention of accumulated water inflow to buildings where SSCs are installed, any such systems are not endangered in case of flooding which can be considered for Temelin NPP location and fulfillment of basic safety functions is confirmed.

A potential beyond design-basis flood could only put at risk those SSC that contribute to the performance of safety functions and are located below the terrain level at the spot height of 507.10 m above sea level (ECW pumping station, DG fuel oil pumps and tanks). Due to the possibility of flooding of the fuel oil pumps for the DGs, including the connection channels that pumps fuel oil from the intermediate tanks to the operating tanks, the long-term operation of DG could be put at risk.

The water inventory for heat removal as a supporting safety function is provided even in case of off-site flooding. In case of extreme off-site floods at Vltava river, it is possible that the Hněvkovice pumping station will be lost for the supply of raw water to compensate for evaporation in case of heat removal to the atmosphere. Even if raw water supply from river is interrupted, there is a sufficient water inventory at site to ensure heat transfer to the ultimate heat sink, ECW cooling pool has a sufficient inventory of water to remove heat from core and from the spent fuel stored in SFP for at least 3x12.5 days (using safety systems only).

3.1.2.2 Main design and construction provisions to prevent flood impact to the plant

Taking into account the potential risks arising only from external flooding caused by extreme rainfalls, all design-basis measures are aimed at ensuring a sufficient resistance of structures against potential floods due to extreme rainfalls.

For the individual structures, the design determines requirements on resistance to accumulated water, which shall ensure that inlet and working openings are designed so that the accumulated water cannot flow to the structures (watertight covers, sufficient height of the opening above the maximum water level), or that sufficient provisional measures against the penetration of accumulated water are installed (beam-type barriers).

In case that ECW pumping station is flooded, ECW pumping stations are equipped with a system of removal of sludge water from the sump at the lowest level (-7.10 m).

3.1.2.3 Main operating provisions to prevent flood impact to the plant

For all floods that are considered on the plant, the performance of fundamental safety functions is not threatened:

a) Reactivity control

b) Heat removal from fuel

c) Confinement of radiation and radioactive materials

Even in case of flooding, reactor shutdown is ensured by the drop of control rods. This function cannot be threatened due to flooding.

Feat removal function cannot also be lost due to floods. Decay heat removal from the reactor in a hot and semi-hot condition would be carried out in the secondary feed and bleed mode by feeding water to SG through the normal or emergency feedwater system (or in another way based on EOPs) and by controlled heat removal to the atmosphere up to the RCS temperature of 150 ^{o}C. After that, heat removal would continue through one of the three trains for decay heat removal to the atmosphere through the ECCS heat exchangers, cooled by ECW, and through ECW cooling pool.

Heat removal from SFP would be ensured through one of the three SFP cooling train through the respective heat exchangers, cooled with ECW, and through ECW cooling pool to the atmosphere.

Barriers against radioactivity leakage - fuel cladding, primary circuit pressure boundary and containment - cannot be simultaneously threatened due to flooding. The containment is a passive system that cannot be threatened by flooding and the isolation of its piping lines penetrations and locks is provided by redundant isolating components, which also cannot fail simultaneously due to flooding.

3.1.2.4 Situation outside the plant, including preventing or delaying access of personnel and equipment to the site.

In case of inaccessibility of the plant due to regional floods the actions to shutdown the reactors and maintain stable state would be would have to be secured by the personnel present on site during the event. Altering would be resolved on an operational basis in cooperation with the state administration bodies (IZS, the army, etc.).

3.1.3 Plant compliance with its current licensing basis

3.1.3.1 Licensee's processes to ensure that SSC needed for achieving safe shutdown state after flood remain in operable condition

The current condition of the plant is in accordance with design requirements. To maintain permanent compliance of the current equipment condition with the design requirements, many regular activities are performed. These activities include:

- Round inspection activities to ensure the required equipment condition and avoid equipment damage, accidents and fires or injuries to people and to ensure equipment operation at a high safety level.
- In-service inspections and tests of the equipment.
- Predictive and corrective maintenance of the equipment.

Inspections and maintenance of rain water sewers are performed regularly. The technical condition of the sewers is inspected once a year and the necessary repairs are provided as necessary. A check of the end shaft at the boundary of drainage from the power plant site for grate cleanliness is carried out once a week.

3.1.3.2 Licensee's processes to ensure that mobile equipment and supplies are in continuous preparedness to be used.

Fire bridage consisting of 16 firemen per shift (non-stop 24 hours service) is available at Temelin NPP. The brigade operates appropriate fire fighting technology and is trained to manage any fire at any place of the location. Fire-suppression equipment and intervention personnel are deployed in a building whose damage in case of a flood is excluded and that would in any other way prevent the use of mobile fire-suppression equipment in case of floods.

Independent means for the transport and pumping of media constitute HZSp's mobile equipment, which is also capable of draining water in case of flooding. For pumping and transport of water the fire brigade has available 4 tank-truck fire engines, 1 combined truck fire truck and 3 trailer fire-engines, with the total nominal capacity of 280 l/s.

3.1.3.3 Potential deviations from licensing basis and actions to address those deviations.

Based on special inspections of compliance of the current condition with the design requirements in terms of resistance to on-site and off-site flooding, carried out after an event in the Fukushima NPP in May 2011, no serious noncompliance were identified between the current condition and the design-basis requirements.

3.2 Evaluation of safety margins

3.2.1 Estimation of safety margin against flooding

The Temelin NPP site has never been, nor it may currently be affected by water floods. The main buildings of Temelin NPP where the nuclear safety related equipment is installed are at the elevation of 507.30 m above sea level, which is 135 m above water level of the Hněvkovice dam on the river Vltava. The safety assessment is provided for Temelin NPP in terms of potential dam failure of water storage reservoirs on the upper reaches of Vltava (Lipno I on the river Vltava and Římov on the river Malše). In case of Lipno I dam failure, the Hněvkovice profile would experience app. 10 000-year flood

During a 10,000-year flood, the Hněvkovice profile would reach a level where most of the pump station used for charging of raw water to Temelin NPP is flooded, thus it would prevent standard operations of raw water supply to Temelin NPP and both NPP units would have to be shut down. The site, however, is provided with sufficient storage volumes to cool down reactors to cold shutdown conditions. During the so-far largest flood on the river Vltava in 2002, the Hněvkovice profile reached a level equal to the maximum elevation considered for this water dam. Water diversion over the dam of the waterworks was done in a standard way and no significant damage was identified either on the pump station for Temelin NPP, or on the waterworks.

Flooding of buildings and structures important to safety by the gravity storm-water drainage system is not possible even during extreme precipitation occurrence. In terms of water drainage, Temelin NPP is built up in a cascade where the nuclear safety related buildings and structures are placed at the highest elevation and descending to the border line of the site, which allows natural gravity drainage even in case of failure of the storm-water drainage system. The civil structures of Temelin NPP are designed to withstand floods under the one-day maximum rainfall where the maximum level of 47,2 mm of 100 year precipitation 88.1 mm of 10,000-year precipitation is reached in the event of complete failure of the drainage system. In addition mobile fire fighting technology is available at the site, customized for drawing off local flooding above the values of 10,000 years maximums.

In view of the fact that stream flooding is inherently excluded and Temelin NPP buildings are designed to be resistant against flooding even in rainfall extremes (watertight covers, height of entry and maintenance openings) at least 100 % margin is provided in comparison to the water level at which water may start to flow into buildings. However, considering gravitation drainage of water from the location such water level can not be reached.

3.2.2 Measures which can be envisaged to increase robustness of the plant against flooding

It is obvious from the previous evaluation that Temelin NPP site is extremely well selected from flooding risk point of view. External flooding on Vltava river may result in loss of Hněvkovice water pumping station and possibility to draw raw water, however under no circumstances it endangers Temelin NPP´s safety. Layout of the design and diversity of SSCs provide sufficient resistance and margins against any design-basis and beyond design-basis flooding.

Any possible adverse consequences of flooding are limited only to fulfillment of supporting safety functions. Possible flooding of diesel oil pumps was identified in case of long term rain, which may affect long term operation of emergency power supply sources (DGs), if at the same time loss of all off-site and on-site AC power sources would occur (see the chapter 5).

The goal of the proposed short-term measures is to increase the robustness of defense-in-depth protection levels in case of flooding. Areas for defense-in-depth improvement are identified in the following table. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation.

Some measures (identified as "PSR finding" in the note) would be implemented regardless of the targeted evaluation, which confirmed efficiency and correctness of previously adopted decision to implement the particular measures for improved resistance of the original design. For detail description of the corrective measures please see the Attachment.

Areas for improvement	**Corrective Action**	**Term** (short-term I / middle-term II)	**Note**
Technical means	DG building enhancement against external flooding	I	PSR finding SOER 2011-1
Emergency preparedness	Capability of OHO to operate out of Emergency Control Center	I	
Procedures/guidelines	EDMG instructions for use of alternative means	II	
Analysis	Analysis of shelters exposer during flooding	II	

4 Extreme weather conditions

For correct understanding of the following text it is necessary to study content of the chapter 1.3, describing plant systems providing main and supporting safety functions of NPP.

4.1 Design basis

4.1.1 Reassessment of weather conditions used as design basis

The load due to natural phenomena is based on the statistical processing of data series for a period of at least 30 years, when these phenomena were measures in the region of NPP or in a region with similar weather conditions. In case of the design-basis load due to climatic effects, the expected repeatability of the respective phenomenon is considered to be once in 100 years. For an extreme computational load due to climatic effects, the expected repeatability is considered to be once in 10,000 years. Structures in seismic class 1 shall withstand the effects of an extreme computational load so that the function of systems essential for nuclear safety is not put at risk. Other structures are subject to design-basis loads.

To rate the resistance of civil structures and equipment against the effects of other natural phenomena, the following extreme climatic effects respective to plant site are taken into account:

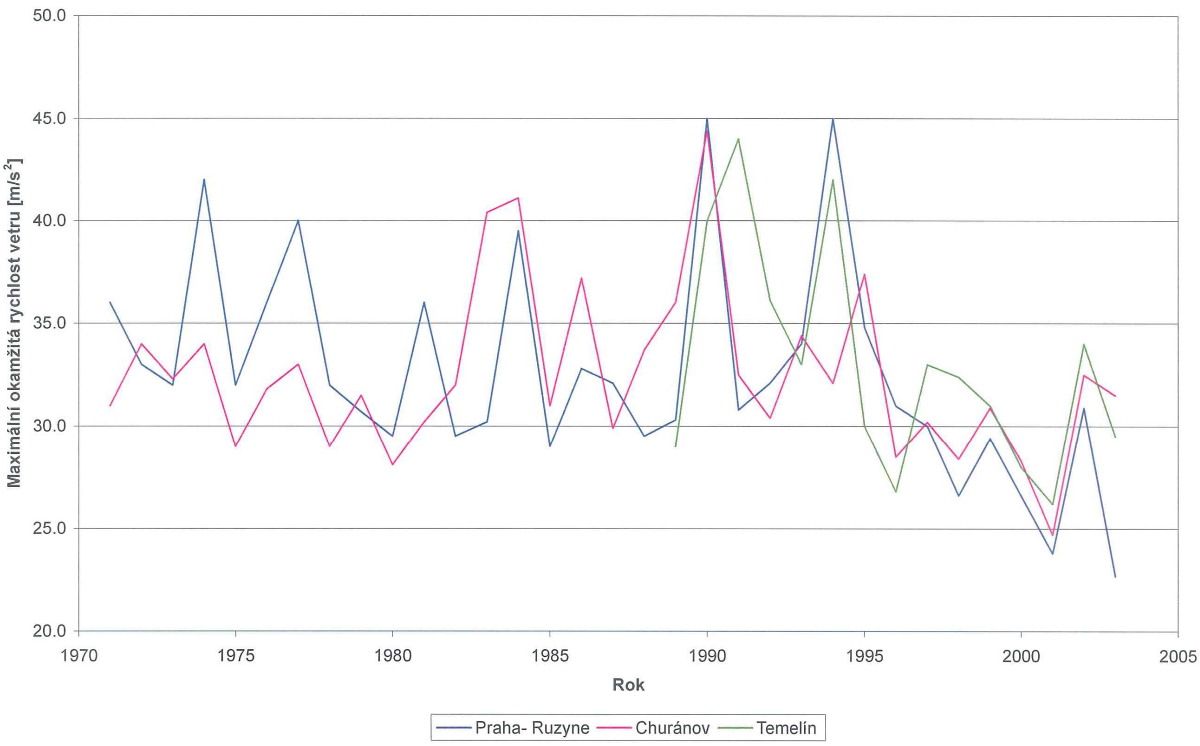
- Wind
- Snow/ice
- High/low temperature

4.1.1.1 High winds

Load determination is based on the measured annual maximum values of immediate wind speed. The input value used for determining wind load is a value specified on the basis of measurements by the Prague-Ruzyně station, i.e. 49 m per second for the frequency of 100 years and 68 m per second for the frequency of 10,000 years.

The wind speed for the upper tornado limit that may occur in the Czech Republic corresponds with the extreme wind speed determined for the frequency of 10,000 years. In terms of load, extreme wind load can be considered as compliant also in terms of load due to tornadoes. As regards flying objects generated by a tornado, it can be assumed that the effects of potential flying objects generated by a tornado are covered by the requirement on the resistance of safety-related structures against the impact of external flying objects.

Fig. 4.1.1-1 Maximum instantaneous wind speed



4.1.1.2 Heavy snowfall and ice

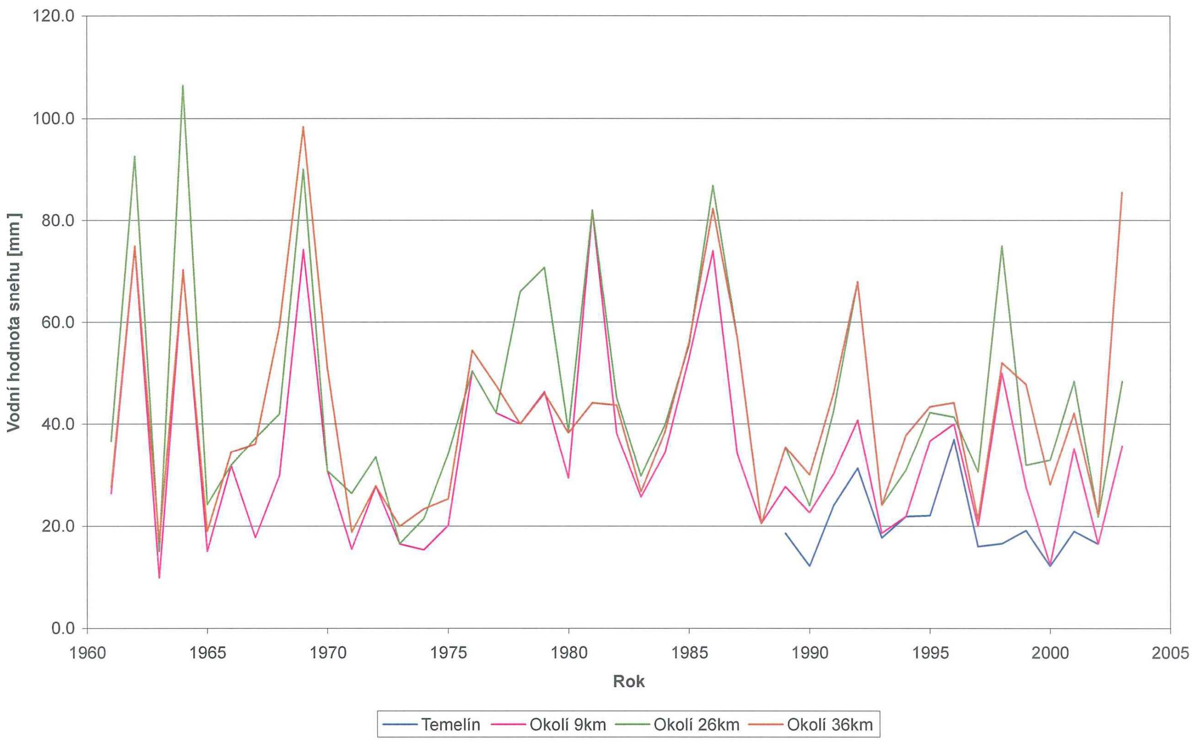
Snow load is expressed as the snow-water content, i.e. the corresponding size of equivalent water column in mm. The value of 92 mm for the frequency of 100 years and the value of 157 mm for the frequency of 10,000 years can be used as the input levels for determining snow and precipitation loads.

Ice cannot be formed in water treatment structures with a free water surface because of their function (heat removal from essential and non-essential consumers), thanks to the temperature of the circulating water.

Since ice formation was observed in the Vltava River up to the confluence with the Lužnice River in the past, the design specified the construction of the Hněvkovice reservoir and the Kořensko reservoir to eliminate the aforementioned occurrences so that water supply and waste water discharge are provided even if ice is formed.

Water supply from the Hněvkovice reservoir is designed so that the covers on the inlets to the pumping station are placed about 8.0 m below the dead dyke level, which provides for smooth water supply under all operating condition.

Fig. 4.1.1-2 Snow water value



4.1.1.3 Maximum and minimum temperature

The loads due to the effects of outdoor temperatures are determined on the basis of measurement of outdoor air temperatures in the stations of Temelín, Tábor and České Budějovice. The value determined on the basis of measurements in the Tábor station is conservatively used as the resulting value. The input values for determining temperature loads are the immediate values of 39.0 ℃ for the maximum annual air temperature and -32.3℃ for the minimum annual air temperature for the frequency of 100 years and 45.6 ℃ for the maximum annual air temperature and -45.9 ℃ or the minimum annual air temperature for the frequency of 10,000 years.

Fig. 4.1.1-3 Maximum air temperature

Fig. 4.1.1-4 Minimum air temperature

4.1.1.4 Consideration of potential combination of weather conditions

SSCs of seismic resistance category 1 were subject to evaluation also with regard to basic and extraordinary combinations of loads. The basic combinations of loads include permanent and random long and short term loads. The extraordinary combinations of loads include permanent loads, random long and short term loads and extraordinary random loads.

For the purpose of extreme natural conditions consideration permanent and long term random loads include, but are not limited to:

- Snow load with lower normal value.
- Temperature climatic effects with lower normal value.

For the purpose of extreme natural conditions consideration short term random loads include, but are not limited to:

- Wind load.
- Snow load with full normal value.
- Temperature climatic effects with full normal value.

For the purpose of extreme natural conditions consideration extraordinary random loads include, but are not limited to:

- Seismic effects.
- Extreme climatic effects occurring one every 10,000 years.
- Load by external impulse pressure wave.

4.2 Evaluation of safety margins

4.2.1 Estimation of safety margin against extreme weather conditions

The safety margins against effect of extreme conditions are given by the difference between values of design and extreme loads (see chapter 4.2.2). In case the design climatic values are reached further operation/shutdown of the units would be considered with regard to prognosis for the future. Taking into account the real achieved values for the climatic conditions on the plant site and the design-basis resistance to extreme conditions, the performance of fundamental safety functions is ensured with sufficient margins in all cases:

a) Reactivity control

b) Heat removal from fuel

c) Confinement of radiation and radioactive materials

Reactor shutdown is ensured by the drop of control rods, which is not anyhow influenced by external conditions.

All systems essential for heat removal are installed in closed (robust) civil structures or in underground structures and their operating media cannot freeze. In case of extreme frost, water freezing in the cooling towers and CNHR can be contemplated. The cooling towers that could freeze after units shutdown do not perform any safety functions,.

If heated water is circulated in ECW cooling pool freezing of ECW cooling pool can not occur even at extremely low external temperatures. The freezing of ECW cooling pool could only occur in case both pumps of any ECW system are shut down for a longer period of time. If the shutdown time during extremely low outdoor temperatures is too long, ECW cooling pool could become frozen through and it could be impossible to re-start ECW pumps. However the Technical Specifications do not allow shutting down all ECW trains in any mode. Even in the reactor shutdown mode at least two ECW trains must be in operation.

Barriers against radioactivity releases - fuel cladding, primary circuit pressure boundary and containment - cannot be simultaneously threatened due to extreme natural phenomena. The containment is a passive system that cannot be threatened by extreme outdoor phenomena and the isolation of its piping lines penetrations and locks is provided by redundant isolating components.

The supplies of media within the plant are provided by piping systems mounted on piping bridges. Since the piping bridges are not in any way secured against external phenomena (earthquake, extreme weather conditions), there are not used for safety functions fulfillment.

In case of an extreme wind, it is highly likely that there would a total loss of off-site power supply for both units (loss of off-site power supply 400 kV and 110 kV), with the ensuing power reduction on both reactors to the level of home consumption. Extreme temperatures can also result in a loss of power supply from main and auxiliary power sources.

In case of a loss of off-site power supply, electrical power supply would be provided from emergency power supply sources, installed in concrete, seismically resistant structures, resistant also to extreme weather conditions. The operating inventory of fuel oil in structures protected against freezing is sufficient for several days of DG operation. However, it is impossible to count on the supply of further fuel oil through the connection pipe installed on service bridges from the plant fuel oil storage facility at a later accident stage (frozen fuel oil, damage due to extreme wind, etc.). Further fuel oil refill would be provided by tank trucks.

For maximum outdoor temperatures, it has been proven by calculations that the maximum design-basis temperature of ECW would be slightly exceeded only for a short period of time in case of a large LOCA. Probability of concurrence of extreme temperatures and LOCA is however extremely low. In other cases the function of the equipment cooled by ECW is not threatened.

For minimum outdoor temperatures, it has been proven by thermal calculations that at extremely low temperatures, ice could be formed on the surface in ECW cooling pool, but it is not an obstacle to ECW cooling pool operation. The inclination of ECW cooling pool walls is so high that it allows ice to move on the surface (rising water level).

4.2.2 Conclusion on the adequacy of protection against extreme weather conditions

All buildings of seismic category 1 can withstand effects of extreme load by natural events. Analyses have demonstrated resistance to the effects of extreme climatic conditions for all SSCs that provide or participate in performance of basic safety functions.

Tab. 4.2.2-1 Specific derived extremes of weather effects

Event (Weather Effect)	Design Level (100-year recurrence)	Extreme Design Load (10,000-year recurrence)
Extreme wind [speed]	49 m/s	68 m/s [1)]
Snow [water column equivalent]	92 mm	157 mm
Maximum temperature [immediate value]	39.0 ℃	45.6 ℃
Minimum temperature [immediate value]	-32.3 ℃	-45.9 ℃

1) Including F2 level tornados

4.2.3 Measures which can be envisaged to increase robustness of the plant against extreme weather conditions

Layout of the design and diversity of SSCs provide sufficient resistance and margins against effects of extreme natural events.

Any possible adverse effects of extreme natural events may lead to shutdown of units, however do not endanger fulfillment of safety functions. Extreme natural events may affect fulfillment of supporting functions, e.g. in case of freezing of media on piping bridges. With regards to the fact that piping bridges are not secured against external events (earthquake, extreme climate conditions) they are not participate on fulfillment of safety functions.

The goal of the proposed measures is to further increase the robustness of defense-in-depth protection levels in case of extreme natural phenomena. Areas for defense-in-depth improvement are identified in the following table. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation. For detail description of the corrective measures see the Attachment.

Areas for improvement	**Corrective Action**	**Term** (short-term I / middle-term II)	**Note**
Technical means	Alternative diesel fuel supply by truck tanks for long term operation of DGs	I	
Personnel	Personnel protection during extreme events	I	
Analysis	Preparation of methodology for evaluation of external effects, verification of completed analysis, possible technical measures	II	

5 Loss of electrical power and loss of ultimate heat sink

For correct understanding of the following text it is necessary to study content of the chapter 1.3, describing plant systems providing main and supporting safety functions of NPP.

5.1 Nuclear power reactors

5.1.1 Loss of electrical power

The electrical system within NPP is designed to meet the requirements of the nuclear island with a respect the properties of electrical grids outside the NPP, taking into account especially the NPP operation safety and electricity production.

Ensuring safety in case of loss of electrical power is settled in the design by a high rate of mutual independence between working and standby home consumption power supply sources as well as by redundancy of the so-called secured power supply systems, which supply safety-related systems and components and have their own emergency sources. Auxiliary supply is provided at the unit level, which prevents the spreading of electrical failures between the units within NPP.

The **main power supply source** for each of the units comprises a pair of transformers, connected to a power output branch from the 1000 MW turbine generator. These transformers can be supplied from two sources, thanks to the use of generator breaker:

- 1000 MW turbine generator (when the unit is operated at power)
- 400kV switchyard Kočín (idle (no power) operation)

This source is available at normal as well as abnormal operation and under emergency conditions, if the link to the 400 kV network or supply from the turbine generator is maintained. If the unit is disconnected from the 400 kV network while the turbine generator is in operation, the supply of home consumption transformers is automatically transferred to the turbine generator by regulating the power down to the level of home consumption supply. The specific case is operation in the "Island" mode when failure in the power supply grid may result in disconnection of certain part of the grid and the turbine-generator may remain connected to this "healthy" part of the grid. The “Island’s” size cannot be determined in advance and can vary a lot – from a large part of the grid to an extreme case of a minimum “Island”, which represents the operation of turbine for the home consumption. The operation of the separate grid mode is described in detail in chapter 1.3.5.

Home consumption transformers supply busses 6kV used for unsecured power supply. Normally, these busses supply busses 6kV for secured power supply, which then supply safety-related systems.

The main power supply sources are not available while the unit is shut down, if regular preventive maintenance is underway on the 400 kV supply system.

An **auxiliary power source** for each of the units comprises a pair of auxiliary transformers, supplied from the Kočín 110 kV switchyard by means of one 110 kV line (a total of two 110 kV lines for both units). The 110 kV network can be supplied in the Kočín switchyard from multiple directions and nodes of the power grid. Auxiliary transformers are connected, by means of auxiliary busses, to 6 kV unit busses for unsecured power supply.

Auxiliary sources are used in normal as well as abnormal operation and under emergency conditions in case of partial or total loss of main power supply sources. Auxiliary sources for both units are mutually backing-up by means of a manual coupling. The auxiliary source is able to back-up the main sources of one unit even while it is partly loaded by the other unit.

Emergency power sources are intended for cases of failure of main and auxiliary sources. Emergency power sources (DGs, batteries) include the safety sources (intended for one unit) with 3x100 % redundancy and common sources (intended for both units) with 100 %+100 % redundancy. Their operability does not depend on the availability of main and auxiliary sources.

Other possibilities of emergency power supply include off-site diverse sources (hydro-alternators of Lipno dam and hydro-alternators of Hněvkovice dam) which were also tested in real situation.

A loss of power supply may affect one or both units of NPP. There is a higher design-basis resistance to the loss of electrical power if the unit is operated at power (sufficient defense-in-depth barriers) than during refueling outages. The least favorable scenario in terms of ensuring safety is the loss of electrical power on both units simultaneously. As regards the possible NPP configuration, the most conservative scenario is a case when one of the units is tripped due to the loss of electrical power and the other unit is under an outage.

5.1.1.1 Loss of off-site power

Design provisions

Loss of off-site power (for example in case of grid decomposition accompanied by loss of both 400 kV and 110 kV switchyards) does not result in automatic transition to emergency sources of power supply in power operation of the units.

If the NPP is disconnected from the external 400 kV network while the unit is operated at power, the turbogenerator 1000 MW TG is able, under design-basis conditions, to regulate down to home consumption power and ensure the power supply of all safety-related systems. If this fails (unit shut down, TG did not operate or did not regulate and failed), it involves a loss of the unit's main power supply sources. In that case, home consumption is automatically switched to auxiliary power supply sources 110 kV (general automatic standby start will take place), DGs will not start, the accumulator batteries are recharged in a standard mode and provide for uninterrupted supply of DC distribution systems.

If the above-mentioned automatic measures do not take place, a LOOP signal is generated from the loss of supply in 6 kV busses of the safety systems. 6 kV safety busses are automatically disconnected from 6 kV busses of unsecured power supply and all three independent safety DGs start and are loaded with emergency load sequencing after connection to corresponding 6 kV safety busses. These safety DGs supply the unit's safety systems and each of the systems is also able to cope with transients in case of a loss of off-site power supply.

At the same time, the loss of power in 6 kV busses of safety-related systems after disconnection from the 6 kV busses of unsecured power supply results in the actuation and sequential loading of the safety related DGs that provide for the supply of equipment relating to safety and equipment essential for the safe run-out of the turbine set. In this mode, the accumulator batteries are recharged in a standard mode and provide for an uninterrupted supply of DC distribution systems.

In off-power unit modes, the main or auxiliary power supply can be inoperable in the long term during regular preventive maintenance.

For detail description of power supply system design see the chapter 1.3.5 and 1.3.6.

In case of a loss of off-site power, the performance of any of the fundamental safety functions is not threatened:

a) Reactivity control

b) Heat removal from nuclear fuel

c) Confinement of radiation and radioactive materials.

In the loss of off-site power mode, units can be maintained in a hot condition in the long term, cooled down to a cold shutdown condition or safely maintained in the outage mode. The power supply for all essential engineering systems and I&C systems is provided if at least one of the three safety DGs on each unit and at least one of the common DG start, but the start of at least one of the three safety DGs on each unit is enough to cool down the unit to a cold condition.

If, during LOOP, the unit is at power, all main coolant pumps are shut down and a reactor trip signal is actuated. Decay heat removal from core is carried out in the natural circulation mode, by removing steam from SGs through a steam dump to atmosphere. Water is fed to SGs using two auxiliary feed water pumps that pump water from the feed water tank; this tank is fed by auxiliary condensate pumps either from the turbine condenser or from 2x800 m^3 demineralized water storage tanks. Alternatively, it is possible to provide water feeding to SGs by means of emergency feed water pumps, which pump water from 3x500 m^3 tanks directly to the selected SG.

If, during LOOP, the unit is in outage, the heat from core is removed by the residual heat removal system. Each of the three cooling circuits includes a circulation pump and a heat exchanger. The heat exchangers are cooled by essential cooling water. The pumps for removing heat from core and the ECW pumps are supplied from safety DGs.

Each SFP section with spent fuel is cooled down by one cooling circuit. Each of the three cooling circuits includes a circulation pump and a heat exchanger. The heat exchangers are cooled by essential cooling water. The SFP cooling pumps as well as the ECW pumps are supplied from safety DGs.

<u>Autonomy of the on-site power sources and provisions taken to prolong the time of on-site AC power supply</u>

In compliance with the basic concept of the nuclear island (3 redundant and independent divisions of safety systems), there are also 3 redundant and independent secured power supply systems available (3x100 %). Each of these power supply systems is a supporting system for the safety systems of the respective division and its preparedness to fulfill the safety functions is tested on regular basis.

- The emergency AC power supply sources of safety systems comprise three independent safety DGs, which are connected to the respective 6 kV safety busses of secured power supply.
- The DC emergency power supply sources are accumulator batteries that are permanently connected to the respective DC busses.

The ensure the necessary level of redundancy, the power supply systems of the safety systems are independent and mutually separated by construction design (separate rooms placed in different parts of the reactor building, building structure with a fire rating of at least 90 minutes), in terms of electrical supply and the control system. Safety power supply systems are designed as seismically resistant, i.e. comprise seismically resistant equipment and are placed in seismically resistant areas. The building structure protects these areas against the effects of the environment during operating and failure events as well as natural events that may occur inside and outside the power plant. The cables for each safety power supply system are placed in separate cable routes, separated from the cable routes of other systems (also with a fire rating of 90 minutes).

In addition to the power supply systems of safety systems, the design also includes two more power supply systems for supplying those parts of the systems relating to nuclear safety and for supplying systems which are not essential in terms of nuclear safety, but which provide for the general safety of people and expensive equipment. These supply systems are designed as two sub-systems that back up each other based on the principle of 100+100 %.

- The AC emergency sources for these systems are two common DGs for both units.
- The DC emergency sources for these systems are accumulator batteries designed specifically for each unit.

If voltage from main or auxiliary power supply sources is available for safety supply systems, the DGs are maintained on hot standby. When a loss of power supply from main and auxiliary sources occurs, the respective DG takes over the supply of the affected part of the distribution system automatically.

Diesel generators have their own fuel oil tanks which are dimensioned, in case of safety DG, for operation at nominal load for at least 48 hours without any fuel oil refill (for a longer period of time in reality). For each common DG at a 100% load (supplying both consumption on both units), the tank is dimensioned to operate for about 12 hours.

With respect to real fuel oil inventory in operating and storage tanks, an operation of safety DG at nominal power is ensured for app. 56 hours. With regard to design of safety systems with redundancy 3x100 % it is possible to gradually use particular safety train and hence to prolong the time of power supply without fuel refilling up to 7 days. All above mentioned times are based on assumption of DG operation at nominal power of 5 MW. The real power of DG (with respect to EOPs actions which require only operation of actual necessary equipment for safe operation) will be app. 2.5 to 3 MW. Such ordinary operating provision can prolong the time of power supply without fuel refilling for additional 40 % up to app. 10 days.

In addition to the tanks installed at the DG, there is also fuel oil storage available on the site with additional fuel inventory 300 m^3 as minimum. Since the fuel oil make-up pumps are supplied from the busses of unsecured power supply, it is necessary to ensure fuel oil refill by mobile means in case of a long-term loss of off-site power. In case of fuel refilling using mobile means it is possible to ensure operation of minimal number of DGs (one safety DG per each unit and one common DG per plant) for additional at least 3 days (with respect to real fuel oil inventory in these tanks of about 1000 m^3 for more than 10 days).

All auxiliary systems of the motor and generator (fuel supply to the motor, lubricating oil, internal cooling circuit, boost air, actuation air) are autonomous and, during DG operation, independent of off-site power supply. Systems that could be affected by a long-term operation of DG (e.g. clogged lubricating oil filters) are redundant subsystems whose one section can be shut down during DG operation, conduct the necessary maintenance, and thus prevent a failure of DG due to a loss of auxiliary systems. The availability of DG and auxiliary systems is regularly tested. The quality of fuel oil is inspected regularly once a month and is maintained in accordance with the applicable requirements.

For detail description of power supply system design see the chapter 1.3.5 and 1.3.6.

5.1.1.2 Loss of off-site power and loss of the ordinary back-up AC power source

Design provisions

In case of loss of main and auxiliary power supply, in parallel with loss of emergency power supply of the unit (safety DG), the following alternative power supply sources are considered to restore power supply:

Internal power sources:

- DGs of non-safety systems (Common DGs of the same design as safety DG).
- Supply from the neighboring unit (this depends on the ability of TG to regulate for home consumption).

External power sources (main strategy for management of loss of AC sources):

- Supply from the Lipno hydro plant (2x60 MW) using dedicated 110 kV distribution lines. In case of grid disintegration, the hydro plant can start up without off-site power supply (blackstart) and, when the line is established from the dispatching room of grid operator, supply power for NPP home consumption. The time needed to bring power from hydro plant to NPP is about 30 minutes and the possibility of this option has been verified in a test (verification of organizational measures to manage an SBO, functionality of physical protection systems, functionality of communications, the roles and procedures for key staff in case of an SBO).

- Hydro plant Hněvkovice – small power source (2x2.2 MW to 2x4.8 MW, depending on water gradient). Power can be brought to NPP through the Kočín 110kV switchyard using a 110 kV auxiliary power supply line.

Availability of both off-site diverse sources of power supply for NPP (hydro-alternators of Lipno and Hněvkovice hydro plants) is assured even in case of grid decomposition, thanks to their capability to start "from black-out". Power from both sources would be supplied to Temelin NPP through Kočín 110kV switchyard on the auxiliary power supply line 110 kV (in case of use of Lipno with proper configuration of 110 kV distribution grid).

There are other AC power supply sources available in NPP, but they are not determined on the basis of the design for supplying the safety systems as part of SBO management.

- DG for supplying turbine lubrication pumps (power 200 kW).
- DG for the data centre (power 1 MW).

Work on restoration of power supply of safety systems from off-site and on-site sources may be conducted in parallel.

The possibility of connecting these sources to the current power supply distribution system has not been solved or tested in the design, their capacity is sufficient for use of these sources for long term charging of accumulators.

Battery capacity, duration and possibilities to recharge batteries

The capacity of the accumulator batteries in power supply systems of safety systems is 3x1,600 Ah. For safety-related systems, the capacity of accumulator batteries is 2x2,000 Ah and 2x2,400 Ah.

In case of a loss of main and auxiliary power sources and DG connection, the accumulator batteries are recharged in a standard mode and provide for an uninterrupted supply of DC distribution systems. . In case of SBO without charging of batteries using diverse sources, the capacity of accumulators is limited by time.

For detail description of the design of electric power supply system, including time limits for accumulator discharging, see the chapter 1.3.5 and 1.3.6.

5.1.1.3 Loss of off-site power and loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources

An event associated with a complete loss of AC electrical power on an unit is a beyond design-basis, highly unlikely accident. It may occur only in case of concurrent failure of all below specified electric power supply system defense-in-depth levels:

- Off-site operating sources - main power supply from 400 kV switchyard,
- On-site operating sources – failure to reject power of turbine home consumption
- Off-site auxiliary sources - back-up power supply from 110 kV switchyard,
- On-site auxiliary sources - power supply from neighboring unit 110 kV switchyard
- All three redundant emergency sources of AC power supply safety systems (safety DGs) of both units
- Both emergency sources of AC power supply for safety related systems (common DGs)
- Off-site diverse sources of AC power supply (hydro-alternators of Lipno and Hněvkovice hydro power plants)

Battery capacity, duration and possibilities to recharge batteries in this situation

In case this mode would occur the accumulators would not be charged (the time until their discharge is several hours depending on current load). Non recovery of power supply and failure to provide heat removal from I&C systems may lead (as a consequence of accumulator discharging) to gradual loss of monitoring of important parameters, control circuits, emergency lighting etc.

Discharge time of the batteries used by the safety systems depends on the progress of current load in time and is expected within several hours. For the minimum discharge times determined by calculation see the chapter 1.3.6.1.

Actions foreseen to arrange exceptional AC power supply from transportable or dedicated off-site source

No alternative or mobile internal AC power sources that could be used to deal with a long-term SBO are available on the site, however there are off-site sources, while their availability and applicability for SBO management has been verified and tested.

Mobile equipment used by HZSp constitutes an alternative means for transporting media. The use of this equipment for technological purposes has not been, however, described. To use this equipment, it is necessary to verify its capacity and the preparedness of connection points that would allow connecting the equipment to SSC for ensuring fundamental safety functions.

<u>Competence of shift staff to make necessary electrical connections</u>

To ensure the performance of safety functions (for design-basis and beyond design-basis scenarios) and to restore power supply in case of an SBO, applicable procedures for the stage preceding core damage are prepared in EOPs and for the stage after core damage are prepared in SAMG. There are also procedures for the restoration of power supply in case of an SBO from on-site sources as well as for using the Lipno hydro plant as an off-site power supply source.

There are two phases of safe unit status recovery after SBO before fuel damage in the core. Initially activities are performed in accordance with the procedure for total loss of 6 kV safety power supply. Activities according to this procedure end at the moment of power supply recovery in at least one train of safety systems. After power supply recovery one the following two procedures are applied for follow-up activities, depending on the existing unit condition. In case that prior power supply recovery no significant changes or disruption of unit parameters occurred (primary circuit temperature and pressure, water level in steam generator etc.) and there is no requirement for operation of ECCS, than such procedure is applied, which leads to stabilization of unit parameters using available systems. In case significant changes in unit condition and parameters occurred prior recovery of power supply to safety bus bars and conditions for activation of ECCS are met, than other procedure is applied, which describes activities with the use of ECCS. Activities for voltage supply from particular selected sources are described in separate procedures.

Plant personnel is sufficiently qualified and trained to use EOPs and SAMG as well as to perform operations to bring power supply from on-site as well as off-site sources in case of an SBO. Within the framework of a shift (IOHO) or POHO, deficiencies concerning the number of staff members necessary to mitigate the consequences of an SBO have not been identified.

In case of an long term SBO, the personnel's orientation would be impaired due to a loss of lighting (only emergency lighting will be functional until the accumulator batteries are discharged). This will extend the time needed for handling operations due to a loss of lighting. In case of an SBO on both units, the shift personnel can be overburdened with activities to restore power supply.

With the shutdown reactor early closure of the containment would need to be provided. Although there are technical means available for containment closing, specific procedures for containment closing in case of SBO with shutdown reactor are not available yet.

In case of total loss of power supply to safety bus bars it is in particular necessary to perform activity preventing loss of primary and secondary coolant. Loss of primary coolant is prevented by closing all discharge paths from the primary circuit. On the secondary side it is possible to limit loss of coolant by maintaining high pressure in the steam generators (by controlled steam discharge from SG to atmosphere), or by closing SG blowdown paths. Heat removal from the RCS must be provided in a way preventing increase of RCS pressure above the opening set value of PRZR relief valve. All these activities can be performed using systems supplied by batteries. With minimization of loss of coolant from primary and secondary circuits it is possible to maintain the unit in hot state for some period of time, without generating an immediate risk for fulfillment of the safety functions.

The TSC manuals deal with the philosophy of possible decrease of DC power supply bus bars load resulting in prolongation of batteries discharging period, and it remains to complete detail procedures describing actions to decrease power of DC power supply bus bars load and use of safety related systems batteries.

According to conclusions of safety systems batteries discharging, in consideration of partial battery load reduction (25% load reduction after 30 minutes), the batteries are capable to supply the loads for more than 4 hours (see chapter 1.3.6.1). With regard to the concept of 3x100 % redundancy safety systems it is possible to gradually deploy individual safety trains and so prolong the period for which power supply is assured to app. 12 hours.

Additional prolongation of safety systems batteries discharging period can be provided by use of safety related systems batteries, whose capacity is comparable to capacity of safety systems batteries.

Knowledge of key operating parameter values is important for SBO management. The values of safety-related variables are communicated through PAMS. The respective I&C systems that send the values of the parameters as well as PAMS as such are supplied from the accumulator batteries. The values of these parameters could be lost either due to a loss of power supply of the respective I&C systems or after a temperature increase in I&C rooms. However the period for which communication of key operating parameter values is provided is always much longer than the period after which heat removal from the core would become endangered.

<u>Time available to provide AC power and to restore core cooling before fuel damage</u>

The most limiting factor in case of an SBO (heat removal from core, heat removal from SFP, loss of I&C system rooms cooling, discharge time of accumulator batteries) is the time for which the unit is able to survive without damage to fuel in core. Another aspect limiting the time when the unit can survive in the case of SBO is the time until the accumulator batteries are discharged. The power supply to essential valves, I&C systems for monitoring the values of essential parameters, the electrical control circuits, emergency lighting etc. is maintained during this time. Accumulator batteries without recharging will ensure power supply for several hours.

As a consequence of an SBO, the I&C of safety systems is jeopardized due to the unavailability of an ultimate heat sink to remove heat losses from the equipment supplied from the batteries. If the cooling of this equipment is not restored, the correct function of the I&C equipment could be gradually affected even if long-term power supply is ensured.

If the unit is operated at power or in hot stand by conditions, the water inventory on the secondary sides of SG would decrease due to a loss of feed water flow to SG in case of an SBO. The SG pressure is controlled by dumping steam in the atmosphere. The SG tubes would be gradually uncovered and the effective heat-exchange surface for heat transfer from RCS would be reduced. This situation would result in a loss of secondary heat removal. As soon as the SGs are not able to remove all decay heat produced in core, the RCS coolant temperature will start increasing. Due to the thermal expansibility of the coolant, this would result in an increase in the PRZR level and therefore a RCS pressure increase.

Until power supply recovery the water inventory in SG enables heat removal from the core via SG to the atmosphere for several hours after SBO occurrence. The limiting condition during SBO is the period after which overheating of the fuel in the core may start. In the worst case the temperature of 650 °C at the core ex it may be reached within several hours after SBO occurrence. Similar period may be available for recovery of power supply in case of loss of heat removal from the core in case of shutdown reactor and reduced coolant level in the reactor, however in this case there is a possibility of gradual gravitation core watering from hydroaccumulators. Overheating of spent fuel in SFP is not threatening within tens of hours after SBO occurrence in case of loss of heat removal from SFP.

Analyses of the SBO scenario, when a loss of secondary heat removal from RCS occurs, imply that without performing alternative activities, described in EOPs, there is a very short time margin to recover heat removal from RCS. The temperature of 650 °C at core exit, which is a limit value in terms of serious damage to the fuel in core, could be, in the most unfavorable case, reached in approximately 2.5 to 3.5 hours from SBO occurrence.

In case of SBO without performing alternative actions, described in EOPs (gradual gravitation watering of the core from hydroaccumulators) during shutdown state of the reactor it may in the worst case result in coolant boiling (middle loop operation with coolant level reduction in the reactor immediately after reactor shut down and cool down to the cold state) in app. 10 minutes after loss of core cooling. Fuel overheating may start in about 30 minutes.

To ensure heat removal, it is necessary to restore power supply of at least one safety busbar by this time, which will prevent the uncovery and potential damage to the fuel at an early stage of the accident.

Due to a loss of electrical power, the cooling of spent fuel is interrupted and water is heated in SFP. The SFP temperature increase rate after an interruption of cooling depends on the initial conditions (time from removal of spent fuel from the reactor, quantity of fuel in SFP etc.). Even at a maximum heat load of SFP, there is no immediate threat of damage to the stored spent fuel after a loss of heat removal from SFP and its damage could occur after tens of hours.

5.1.1.4 Conclusion on the adequacy of protection against loss of electrical power

The power supply sources at Temelin NPP provide a sufficient design robustness as well as safety assurance level in the event of loss of off-site power. They are designed with a high level of mutual independence of working and standby house load sources, and further with redundant essential power supply systems that supply safety related systems and components and dispose of their own emergency power sources (DG and accumulator batteries). The house load power supply is designed per each unit to prevent on-site propagation of electrical faults between the units.

Higher design resistance against loss of power is ensured during power operation of the reactor (sufficient defense-in-depth barriers) than during refueling outages. The worst scenario in terms of safety assurance is loss of electrical power to both units at the same time.

There are a total of 8 emergency alternate power sources available on the site (3 safety DGs per unit and 2 common DGs for both units). In the mode of loss of off-site power, Temelin NPP reactors can be maintained in long-term fail-safe state, or cooled down to a cold shutdown state, or safely maintained in the outage mode (power supply is provided to all the necessary mechanical systems and I&C systems) if at least one of the DGs has started up in each unit. For each DG, diesel oil inventory is provided for more than 2 to 3 days, without a need for external refueling. On-site additional diesel oil inventory is available to further extend the DG operation.

If off-site power supply is lost and TG is not regulated to home consumption, the supply of safety systems, safety-related equipment and equipment for the safe run-out of the turbine set is provided by emergency AC power supply sources (DGs) and by emergency sources of uninterrupted DC power supply (accumulator batteries). When the respective DG is operated, the accumulator batteries are continuously recharged. In case of long term operation of emergency power supply sources it would be necessary to supply diesel fuel by mobile means.

In case of a total loss of AC power (SBO), in case of concurrent failure of all below specified electric power supply system defense-in-depth levels:

- Off-site operating sources - main power supply from 400 kV switchyard,
- On-site operating sources – failure to reject power of turbine home consumption
- Off-site auxiliary sources - back-up power supply from 110 kV switchyard,
- On-site auxiliary sources - power supply from neighboring unit 110 kV switchyard
- All three redundant emergency sources of AC power supply safety systems (safety DGs) of both units
- Both emergency sources of AC power supply for safety related systems (common DGs)
- Off-site diverse sources of AC power supply (hydro-alternators of Lipno and Hněvkovice hydro power plants),

the only sources for supplying the safety systems and safety-related systems are emergency sources of uninterrupted DC power supply (accumulator batteries). When the respective DG is not operated, the accumulator batteries are not recharged and the time until their discharge is several hours, depending on the current load. Significant prolongation of the discharging period can be achieved by controlled reduction of batteries load by gradual deployment of individual trains and use of safety related systems batteries, which have a high capacity. Alternatively other AC sources may be used for long-term charging of batteries, which are available on the site.

5.1.1.5 Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power

In case of total loss of the alternating current power supply units (SBO), only the direct-current uninterrupted emergency power supply systems (accumulator batteries) are the sources of power supply for the safety systems and the safety-related systems. Permanent provision of fundamental safety functions in SBO mode is dependent on recovery of AC power supply, for which several options exist in the existing design.

Although multiple failure of defense-in-depth of NPP´s electric systems would have to occur before SBO, considering severity of SBO consequences additional measures have been proposed in order to improve even so robust design of safety systems, including possibility of connection of alternative sources to existing power supply and their testing.

The goal of the proposed short-term measures is to eliminate the identified risks by strengthening defense-in-depth protection levels in case of initiation events beyond the framework of the current design (earthquake, floods, extreme conditions, results of human interference, etc.) that could result in a loss of the ability to perform the safety functions during an SBO:

1. Propose and implement alternative means for AC power supply of the existing equipment to ensure cooling and heat removal from core and SFP, including the possibility of their connection to the existing power supply systems.
2. Propose and implement diversified means for cooling and heat removal from core and SFP, including the possibility of their connection to existing technology.
3. Propose and implement alternative means to ensure DC power supply and cooling of I&C systems essential for status monitoring and controlling selected components.
4. Propose and implement alternative means for the activities and functional communication (on-site and off-site) of personnel.

Specific areas for defense-in-depth improvement are identified in the table of the chapter 5.1.3.3. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation.

5.1.2 Loss of the ultimate heat sink

5.1.2.1 Design provisions to prevent the loss of the primary UHS

The ultimate heat sink for is the ambient atmosphere. Heat not utilized in the unit's operation at power and the decay heat from core after reactor shut down, can be discharged to the ultimate heat sink in several operating ways:

- By heat removal through the main turbine condenser system to the circulating water and to the atmosphere through the cooling towers - in case of normal and abnormal operation at power, start-up and shut-down of TG and in emergency conditions after reactor trip if main and auxiliary power supply sources are provided.
- Removal of decay heat from core and safety system components using an essential cooling water system to ECW cooling pool, and then to the atmosphere - during normal

and abnormal operation and accident conditions, with the possibility to bring the reactor to a cold shutdown conditions.

In case operating methods of heat removal to the ultimate heat sink are unavailable, alternative methods of heat removal can be applied:

- Direct removal of heat by discharging steam from SGs to the atmosphere, with simultaneous supply of feed water during abnormal and emergency operation; this option facilitate long term residual heat removal from core but does not make it possible to bring the reactor to a cold shutdown conditions (cooling to about 110 ^{o}C).
- An alternative "feed and bleed" method (controlled discharge of the coolant from RCS to the containment, removal of heat through ECCS heat exchangers to the ECW system and injection of the cooled coolant to RCS by means of ECCS pumps) – only under accident conditions if it is impossible to use secondary heat removal.

Heat removal from RCS through the SG secondary side is ensured when the reactor is shut down while the unit is maintained in a hot condition or in the first stage of unit cool-down. It is ensured by feed water flow to SG (normal or emergency feed water system) and removal of steam from SG to the main condenser or to the atmosphere. Heat removal from main condensers to the circulating water is not evaluated herein because it may not be available (these systems are not safety-related).

The necessary feed water flow to each SG can be ensured by the auxiliary feed water system (redundancy 2x100 %) as well as by the emergency feed water system (redundancy 3x100 %). The residual heat from core can be removed by each of the four steam dumps to the atmosphere, installed on inseparable parts of main steam pipelines. It is possible to remove heat from RCS to the atmosphere up to the temperature of about 110 ^{o}C; for lower temperatures, this removal is not effective.

The essential cooling water system, which transfers heat through ECW cooling pool to the atmosphere as the ultimate heat sink, is used for RCS cooling down to a cold shutdown conditions, removing heat from the spent fuel in SFP and removing heat from the components of safety systems and safety related systems. Three ECW systems are in operation at the same time (redundancy 3x100 %). From each ECW system, heat is removed to a separate ECW cooling pool, where it is transferred to the atmosphere through water evaporation from a water surface and from water sprayed using nozzles. Each ECW cooling pool comprises two functionally independent halves, where one half of the ECW cooling pool is in operation, while the other half is empty and is used as a back-up.

To remove heat from the spent fuel placed in SFP, it is possible to use alternatively the containment spraying system pumps for make-up of SFP through the dedicated line for this purpose and remove the heat by evaporation to the containment.

The ultimate heat sink cannot be lost. The transfer of residual heat from core, SFP and safety system equipment to the ambient atmosphere as the ultimate heat sink is based on the passive physical principle of heat transfer from an auxiliary medium to the atmosphere. A loss of the ultimate heat sink can be therefore only evaluated as a loss of the heat transfer ability, i.e. a loss of the function of the systems that provide for the flow of the heat transfer media between the sources of heat and the atmosphere. For the purposes of evaluation of a loss of the ultimate heat sink, it is possible to consider a loss of the systems for heat removal through the secondary circuit and a loss of the ECW system. Since heat removal through SG can be only used to reduce the RCS parameters to values suitable for bringing the residual heat removal system into operation, when it continues transferring heat to the ECW system, a loss of the ultimate heat sink for the purposes of this evaluation shall refer particularly to a loss of the ability of the ECW system to transfer heat from core, SFP and safety system equipment to the ambient atmosphere.

This approach is based on the fact that heat removal to the atmosphere directly from SG using the secondary feed and bleed method can be provided by a combination of several options, diversified in terms of solution and redundant in terms of system configuration. The systems that contribute to heat removal are also available in case of operation of only emergency power supply sources. In addition, if heat removal through the secondary circuit is lost, EOPs describe the possibility of emergency heat removal using the primary feed and bleed method.

Due to the unit-based configuration of residual heat removal systems (without any possibility of interconnecting them), a loss of the heat removal function is evaluated separately for each unit. A loss of the ultimate heat sink results in smaller risks for heat removal from core in modes in which the reactor is closed (all unit operating modes, except for a refueling outage), thanks to the possibility to remove heat through SG. An outage mode in which the reactor is not closed and heat removal through SG is not effective results in similar risks for fuel placed in the reactor as well as for fuel placed in the spent fuel storage pools.

Water supply for NPP´s technology is provided from Hněvkovice water reservoir immediately by the dam body on the left shore. The pumping station is designed for supply of 1.3-4.16 $m^3.s^{-1}$ provided that it is capable to provide the volume of 4.16 $m^3.s^{-1}$ by operation of 1-4 pumps (the capacity is designed for the original design 4 x 1000 MW). Other two pumps present a 50% margin. 6 vertical pumping systems are installed in the pumping station.

Water from the pumping station is transported to 2x15.000 m^3 head reservoir on the site by two underground DN 1600mm steel pipes delivery conduits app. 6.2 km long. Capacity of both conduits is 4.16 $m^3.s^{-1}$, in case of failure of one of them the other is capable to provide the guaranteed volume of 3.4 $m^3.s^{-1}$ with concurrent operation of 4 pumps. Power supply of the pumping station is provided by double 110 kV line from Kočín 400/110 kV switchyard to 110/6 kV switchyard and transformer within the pumping station. Information on operating and failure conditions of the pumping station is transmitted by communication cable to the on-site water management control room.

For detail description of heat removal from the core and SFP to the ultimate heat sink see the chapter 1.3.

5.1.2.2 Loss of the primary ultimate heat sink

Availability of an alternate heat sink

Calculations have proven that one ECW cooling pool is able to remove all heat from the unit with a shut-down reactor without make-up in a long-term setting even in the most unfavorable condition, without the ECW temperature significantly exceeding the maximum design-basis value. The most unfavorable case is a LOCA on one unit and shutdown of the other unit, i.e. the heat source for ECW is maximal. Thanks to the existence of three redundant ECW systems it is possible to establish that heat removal to the ultimate heat sink can be secured at least for 30 days without any make-up on condition that all safety divisions are gradually used or that the inventory of water from ECW cooling pool of operable ECW systems is pumped using mobile devices to ECW cooling pool of an operable ECW system.

Taking into account the redundancy of ECW systems 3x100 % and another internal redundancy of pumps for each ECW train 100+100 %, a loss of the heat transfer ability from the sources to ECW cooling pool is conditional on the inoperability of all ECW pumps (a total of 6 pumps). Because of the spatial separation of the systems and pumps, the independence of power supply and other supporting systems, the simultaneous inoperability of all ECW pumps is extremely unlikely. Even if only one pump is operated in a single train of the ECW system, it is possible to ensure the performance of basic safety functions. The only possible causes for a loss of all ECW pumps could be external flooding or SBO.

Even in case of total loss of ECW in the hot state, heat removal from the core can be provided by normal operation systems, which are independent on operation of ECW system - by SG feeding by auxiliary feed water pumps and steam dump to the condenser or to atmosphere.

The key non-technology means to be used in case of a loss of the ultimate heat sink include the pumping equipment of HZSp. This equipment, however, has not been considered for use to mitigate the consequences of technological failures yet. In addition to this equipment, there are no other alternative or mobile sources in NPP to ensure circulation or heat removal from ECW heat sources that could be used to solve a loss of the ultimate heat sink.

Calculations have proven that one ECW cooling pool is able to remove all heat from both units for 12.5 days, without any refilling. To meet the requirement on ensuring heat removal for at least 30 days, it is necessary to re-pump the water inventory from ECW cooling pool of the inoperable ECW systems, using mobile means, to ECW cooling pool of an operable ECW system. A feasibility analysis of mobile fire fighting equipment implies that it is possible to pump water between ECW cooling pool using such means.

Possible time constraints for availability of alternate heat sink and possibilities to increase the available time

A loss of heat transfer to the ultimate heat sink is associated with the inability of unit cool-down. With a stable or increasing RCS temperature, no positive reactivity is inserted in, i.e. it is not necessary to compensate the addition of positive reactivity. While the units remain in a hot condition, the reactivity management function is preserved.

Since the operability of the ECCS systems, containment spraying, emergency feed water to SG systems etc. depends on the operation of ECW systems, they cannot be contemplated for use to deal with the given condition. Heat removal from core would be ensured by normal operation systems, which do not depend on the operation of the ECW system - by feeding SG using auxiliary feed water pumps, steam removal to the condenser or the atmosphere. In hot conditions, secondary feed and bleed mode, could be used to remove heat from RCS in the long term.

A different situation occurs if a unit is in an outage mode (while the reactor is opened), when heat removal from core depends on the operation of ECW. A loss of ECW results in a core temperature increase. In that case, it is possible to fill the wet transport pools. Without heat removal, the temperature in the wet transport pools will increase up to the saturation level. If evaporation is compensated by make-up, it is possible to remove heat in this condition in the long term.

In case of a loss of ECW, there is the same situation in terms of heat removal from SFP as in case of an SBO, i.e. the cooling of spent fuel is interrupted and the water in SFP is heated. The temperature increase rate in SFP after an interruption of cooling depends on the initial conditions (time from removal of spent fuel from the reactor, quantity of fuel in SFP etc.). Even at a maximum heat load of SFP, there is no immediate threat of damage to the stored spent fuel after a loss of heat removal from SFP and its damage could occur after tens of hours in a late stage of the accident.

Under normal operation and abnormal conditions, the heat from the containment is removed using air-conditioning ventilation systems cooled by ECW. If temperature increases, heat removal from the containment can be provided using the chilled water system. If a long-term loss of ECW occurs and the chilled water system is not operable, there will be a loss of heat removal from the containment. Temperature in the containment will start growing. The supply of cooler air from the outside will be preserved and underpressure will be maintained by systems that ensure air exhaust from the containment.

If the containment is closed, the containment integrity could be threaten only at a later stage of the accident, taking into account its design and ability to withstand thermal and pressure effects. If the containment is opened (especially if the reactor is opened), radioactive releases from CNTN could occurred while the temperature is maintained at the saturation level leak because there are no procedures for the timely closing of the containment.

The functionality of I&C systems and the knowledge of the values of the key parameters are essential for ensuring safety functions. The respective I&C systems as well as PAMS as such will be affected by a temperature increase in I&C rooms after a loss of ECW. Heat removal from I&C rooms of the safety systems can be alternatively provided using the non-essential service water system. This possibility is described and used, as a standard, for planned ECW outages and increases resistance in the performance of safety functions if ECW is lost.

5.1.2.3 Loss of the primary ultimate heat sink and the alternate heat sink

<u>External actions foreseen to prevent fuel degradation</u>

Mobile equipment used by HZSp constitutes an alternative means for transporting media. The use of this equipment for technological purposes has not been, however, described. To use this equipment, it is necessary to verify its capacity and the preparedness of connection points that would allow to connect the equipment to the technology for ensuring fundamental safety functions.

<u>Time available to recover one of the lost heat sinks or to initiate external actions and to restore core cooling before fuel damage</u>

If the unit is operated at power or in a hot condition, it is possible to remove heat from core, in case of a loss of UHS, by normal operation systems, which do not depend on the operation of the ECW system - by feeding SG using auxiliary feed water pumps, steam removal to the condenser or the atmosphere. In a hot condition the secondary feed and bleed mode could be used to remove heat from RCS in the long term.

A different situation occurs if a unit is in an outage mode (while the reactor is opened), when heat removal from core depends on the operation of ECW. A loss of ECW results in a temperature increase in core. In that case, it is possible to fill the wet transport pools. Without heat removal, the temperature in the wet transport pools will increase up to the saturation level. If evaporation is compensated by make-up, it is possible to remove heat in this condition in the long term.

From a long-term point of view, it is necessary to recover the operation of the ECW system in at least one safety division, which allows unit cool-down to a cold shutdown conditions.

Water level presents another important aspect which significantly affects the period after which the saturation temperature in SFP is reached. Level drop in SFP below 754 cm leads to loss of circulation through SFP cooling system and level drop below 550 cm results in exposure of stored fuel assemblies. With the loss of the last functional redundancy of the heat removal system from SFP and achievement of the saturation temperature a significant "cliff edge" effect occurs, when it is necessary to remove heat by coolant boiling in SFP and its evaporation to the containment.

In case the maximum residual power output is produced in SFP (the all fuel from the core is in SFP and remaining part of SFP is filled with spent fuel from previous fuel cycles) the minimum period until reaching the saturation temperature is app. 30 hours. Volume of water in SFP in the fuel storage mode is 223 m^3 in sections 01 and 03 each, and 104 m^3 in the section 02 (the volume is approximately double in the refueling mode). With regard to the above specified volumes of individual SFP sections the available coolant inventory in the pit provides two times prolongation of the period until reaching the saturation status (app. 60 hours) and the available inventory of coolant in the storage tanks for refueling provides four times prolongation of this period until reaching the saturation status (app. 120 hours).

Due to a loss of UHS, the cooling of spent fuel is interrupted and water is heated in SFP. The temperature increase rate in SFP after an interruption of cooling depends on the initial conditions (time from removal of spent fuel from the reactor, quantity of fuel in SFP etc.). Even at a maximum heat load of SFP, in case of SFP make-up to compensate evaporation there is no immediate threat of damage to the stored spent fuel after a loss of heat removal from SFP. Without SFP make-up to compensate evaporation damage of stored spent fuel could occur after tens of hours.

As a consequence of a loss of UHS, the I&C of safety related systems is jeopardized due to the impossibility to remove heat losses from equipment supplied from accumulator batteries.

If the cooling of this equipment is not restored, the correct function of I&C equipment could be gradually affected.

5.1.2.4 Conclusion on the adequacy of protection against loss of ultimate heat sink

The ultimate heat sink is the surrounding atmosphere. Heat unused during the power operation of the unit, or residual heat after the reactor shutdown can be removed to the ultimate heat sink - the atmosphere - in several ways: Heat transfer between safety essential heat sources and the atmosphere is provided by ECW system through ECW cooling pool.

Water inventory available in ECW cooling pool is sufficient for app. 30 days of ECW system operation for residual heat removal from shutdown reactors without external water supply to ECW system. In total 6 ECW pumps are available per each unit. With respect to the spatial separation of the systems and the pumps, the autonomy of the power supply as well as other auxiliary systems, the concurrent unavailability of all the ECW pumps is highly unlikely. Even in case of operation of only one pump in one train of the ECW system, it is possible to cover the fundamental safety functions.

Considering existence of several operating methods of heat removal and several alternative methods in case of their unavailability, even in case of highly improbable loss of the ultimate heat sink in case of a loss of the ability of the ECW system to transfer heat from core, SFP and safety system equipment to the ambient atmosphere it would be possible to maintain the unit in a hot or semi-hot condition in case of a loss of ECW by using direct heat removal to the atmosphere through SG, which is independent of heat removal using the ECW system. This option provides sufficiently long time for preparation of alternative methods of heat removal. A loss of ECW is always associated with the impossibility to cool the unit down to a cold condition and maintain it in a cold condition in the long term.

If neither operating nor alternative means for heat removal are available the consequences of a loss of the ability of heat removal to the ultimate heat sink can be as follows:

- Damage to the fuel in core and the spent fuel stored in SFP due to a lack of alternative ways of heat removal from core, SFP and components cooled by ECW.
- A loss of cooling of emergency AC power supply sources in case of a LOOP can result in a SBO.
- If the containment was not sufficiently isolated during an outage, radioactive releases to the surroundings could occur.

- A loss of the ability to control the systems and components and to monitor the values of important parameters due to a loss of functionality of I&C systems in case of inability to remove heat losses from I&C equipment.

5.1.2.5 Measures which can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink

Although multiple failure of defense-in-depth would have to occur before the total loss of capability to transfer heat to the ultimate heat sink, considering severity of consequences of such conditions additional measures have been proposed in order to improve even so robust design with regard to heat transfer to atmosphere as the ultimate heat sink.

The goal of the proposed short-term measures is to eliminate identified risks by strengthening defense-in-depth protection levels in case of initiation events beyond the framework of the current design (earthquake, floods, extreme conditions, results of human interference, etc.) that could result in a loss of the ability to perform the safety functions during a loss of UHS.

1. Propose and implement diversified means for cooling and heat removal from core and SFP, including the possibility of their connection to existing technology.
2. Propose and implement alternative means to ensure the cooling of I&C systems essential for status monitoring and controlling selected components.
3. Describe and use alternative and diversified means (designed under points 1 and 2) – emergency plans (EDMG), with the goal to ensure cooling and heat removal from core and SFP.

For areas for defense-in-depth improvement in case of events which may result in loss of capability to fulfill safety function see the table in the chapter 5.1.3.3. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation.

5.1.3 Loss of the primary ultimate heat sink, combined with station black out

5.1.3.1 Time of autonomy of the site before loss of normal reactor core cooling condition

<u>Loss of the primary ultimate heat sink, combined with station black out</u>

Only one ultimate heat sink for removing heat from safety systems exists - the ambient atmosphere. The ECW system is a means for transferring decay heat from core, from the spent fuel stored in SFP and from components of safety systems to the ultimate heat sink.

In case of an SBO event, the ECW pumps are not supplied. Since ECW is the medium for heat transfer to the atmosphere, the forced removal of heat from RCS and SFP to the atmosphere will be lost along with the SBO. An SBO event automatically means a loss of the ultimate heat sink for the given unit due to a loss of electrical power to ECW pumps.

In case of a loss of the ultimate heat sink and a simultaneous loss of electrical power from main and auxiliary sources, an SBO situation on the given unit will occur due to a loss of cooling of DG. This is because of the mutual dependence between DG and ECW - a failure of either will result in a loss of both.

Heat removal through the secondary circuit (SG) could be until secondary water inventory is lost used in a hot and semi-hot condition of the unit. However, there is no back-up means for heat removal from the spent fuel stored in SFP.

The information above implies that the operability of the ECW system for heat transfer to the ultimate heat sink and the operability of emergency power supply sources are interconnected.

<u>Loss of UHS as a consequences of SBO</u>

The ECW system pumps that provide for heat transport from the sources to the ultimate heat sink are supplied from secured power supply. In case of an SBO, ECW is also always lost. Taking into account the possibility of heat removal from SG directly to the atmosphere, this condition does not mean an immediate loss of the ultimate heat sink. In case of an SBO, the heat from core can be removed also using the limited water inventory in SG. The ability to remove heat from the spent fuel in SFP would be lost at a late stage of the accident.

A loss of ECW in case of an SBO, however, limits the time of availability of the values of important parameters. The heat losses from I&C equipment supplied from accumulator batteries without a cooling function due to the unavailability of ECW systems will result in a temperature increase in I&C rooms and a subsequent loss of the respective I&C systems.

<u>SBO as a consequences of loss of UHS</u>

A loss of the ultimate heat sink by itself will not affect the plant home consumption power supply if power supply from main or auxiliary sources is provided. If off-site power supply is lost and the turbine generator is not regulated to home consumption power supply while ECW is not available, the emergency AC power supply sources (safety DG) will be actuated, but when they are connected to distribution system and loaded, they will gradually fail due to overheating because of a loss of ECW cooling.

The power supply of safety systems will be provided only by the operated accumulator batteries. Because it is impossible to remove the generated heat from I&C systems, temperature will grow in the rooms, I&C systems will be gradually lost and there will be a loss of monitoring of the values of important parameters.

5.1.3.2 External actions foreseen to prevent fuel degradation

Mobile equipment used by HZSp constitutes an alternative means for transporting media. For pumping and transport of water the fire brigade has available 4 tank-truck fire engines, 1 combined fire truck and 3 trailer fire-engines, with the total nominal capacity of 280 l/s.

The use of this equipment for technological purposes has not been, however, described. To use this equipment, it is necessary to verify its capacity and the preparedness of connection points that would allow connecting the equipment to the technology for ensuring fundamental safety functions.

5.1.3.3 Measures, which can be envisaged to increase robustness of the plant in case of loss of primary ultimate heat sink, combined with station black out

The functions of power supply from emergency sources and heat transfer to the ultimate heat sink are closely related and loss of one of the functions may affect the other function and vice versa. Although multiple failures of defense-in-depth levels would have to occur before the loss of both functions, considering severity of consequences of such conditions additional measures have been proposed in order to improve even so robust design with regard to heat transfer to atmosphere as the ultimate heat sink. The measures to enhance the robustness of the units in case of SBO combined with the loss of UHS are the same as the measures identified in case of SBO in chapter 5.1.1.5 and in case of the loss of UHS in chapter 5.1.2.5

The goal of the proposed short-term measures is to eliminate identified risks by strengthening defense-in-depth protection levels in case of initiation events beyond the framework of the current design (earthquake, floods, extreme conditions, results of human interference, etc.) that could result in a loss of the ability to perform the safety functions during an SBO in combination with a loss of UHS.

See the following table for areas for defense-in-depth improvement in case of events which may result in loss of capability to fulfill safety functions. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation. For detail description of the corrective measures see the Attachment.

Areas for improvement	**Corrective Action**	**Term** (short-term I / middle-term II)	**Note**
Technical means	Alternative water feeding to SG/SFP/RCS (with unsealed RCS)	II	
Technical means	Alternative source for batteries charging and power supply of selected appliances	I	
Technical means	Alternative diesel fuel supply by truck tanks for long term operation of DGs	I	
Analysis	Analysis of heat removal from I&C systems after loss of ECW	I	
Technical means	Switching over of CNTN isolation valves of HVAC to batteries	II	
Procedures / guidelines	Use of safety DGs of the other unit during SBO	I	
Analysis	Analysis of batteries discharging period in case of controlled load reduction, completion of procedures	I	
Procedures / guidelines	Procedure for containment isolation in shutdown state	I	SOER 2010-1
Analysis	Heat removal from SFP without make up	I	
Procedures / guidelines	Procedure for power supply recovery after SBO of all units	I	
Personnel	Provision of enough personnel in case of long term SBO	I	
Analysis	Analysis of shift personnel capabilities in case of SBO of both units	I	

Areas for improvement	**Corrective Action**	**Term** (short-term I / middle-term II)	**Note**
Procedures / guidelines	EDMG instructions for use of alternative means	II	
Communications	Alternative communication means after seismic event	I	
Procedures / guidelines	Procedure for operation of units with long-term power supply from emergency sources	I	

5.2 Spent fuel pool storage pools

Considering design of VVER1000 with SFP situated inside the containment just next to the reactor, similar means are used for power supply of systems and heat removal from spent fuel in SFP to the ultimate heat sink as for heat removal from the core. Therefore evaluation applicable to loss of power supply and loss of UHS for heat removal from SFP is provided in the appropriate sections of the chapter 5.1.

Although multiple failure of defense-in-depth would have to occur before the total loss of capability to remove heat from spent fuel in SFP, considering severity of consequences of such conditions additional measures have been proposed in order to improve even so robust design with regard to heat removal from SFP to the ultimate heat sink as a consequence of SBO or loss of UHS.

The objective of the proposed measures is strengthening of defense-in-depth levels in case of initiation events beyond the scope of the current design (earthquake, floods, extreme conditions, results of human interference, etc.) that could result in a loss of the ability to perform the safety functions during an SBO.

For areas for defense-in-depth protection improvement in case of events which may result in loss of capability to fulfill safety function see the table in the chapter 5.1.3.3. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation.

6 Severe accident management

For correct understanding of the following text it is necessary to study content of the chapter 1.3, describing plant systems providing main and supporting safety functions of NPP.

6.1 Organization and arrangements of the licensee to manage accidents

6.1.1 Organization of the licensee to manage the accident

The basic objective of NPP safety is to prevent uncontrolled release of radioactive materials, in particular those generated in the reactor core. To achieve this goal the design is based on so called "defense-in-depth" concept, which is based on the principle of use of multiple physical barriers preventing leakage of radioactive materials. The objective of severe accident management is to provide 4th level of defense-in-depth (mitigate consequences of severe accident), after failure of 3rd level of defense-in-depth (i.e. failure to prevent damage of fuel during design-basis and beyond design-basis events). Accident management is related to emergency preparedness system whose main objective is provision of 5th level of defense-in-depth (mitigation of radiation consequences of significant radioactive material leakage).

The accident management and emergency preparedness are provided by means of a robust set of measures having personnel, administrative and technical character. In the personnel area, there is the Emergency Response Organization and the provision of activities pertaining to individual positions; in the administrative area, there is the implementation of the respective procedures, manuals and instructions combined with the use of the technical support centers (from which the staff provide for control and execution of interventions) and in the technical area, there is the provision of functional state of the required scope of technical means to implement the respective strategies. Execution of interventions upon occurrence of accident events is secured, in the first (prevention) stage, always by means of the continuous shift operation staff. In case the scope of the event exceeds the possibilities of the continuous shift operation staff, the second stage (extenuation of consequences) starts, when the Emergency Response Organization is activated. In this case, the Emergency Commission supported by the Technical Support Centre takes over the responsibility for the interventions management.

Fig. 6.1.1-1 Relation between physical barriers and defense-in-depth levels

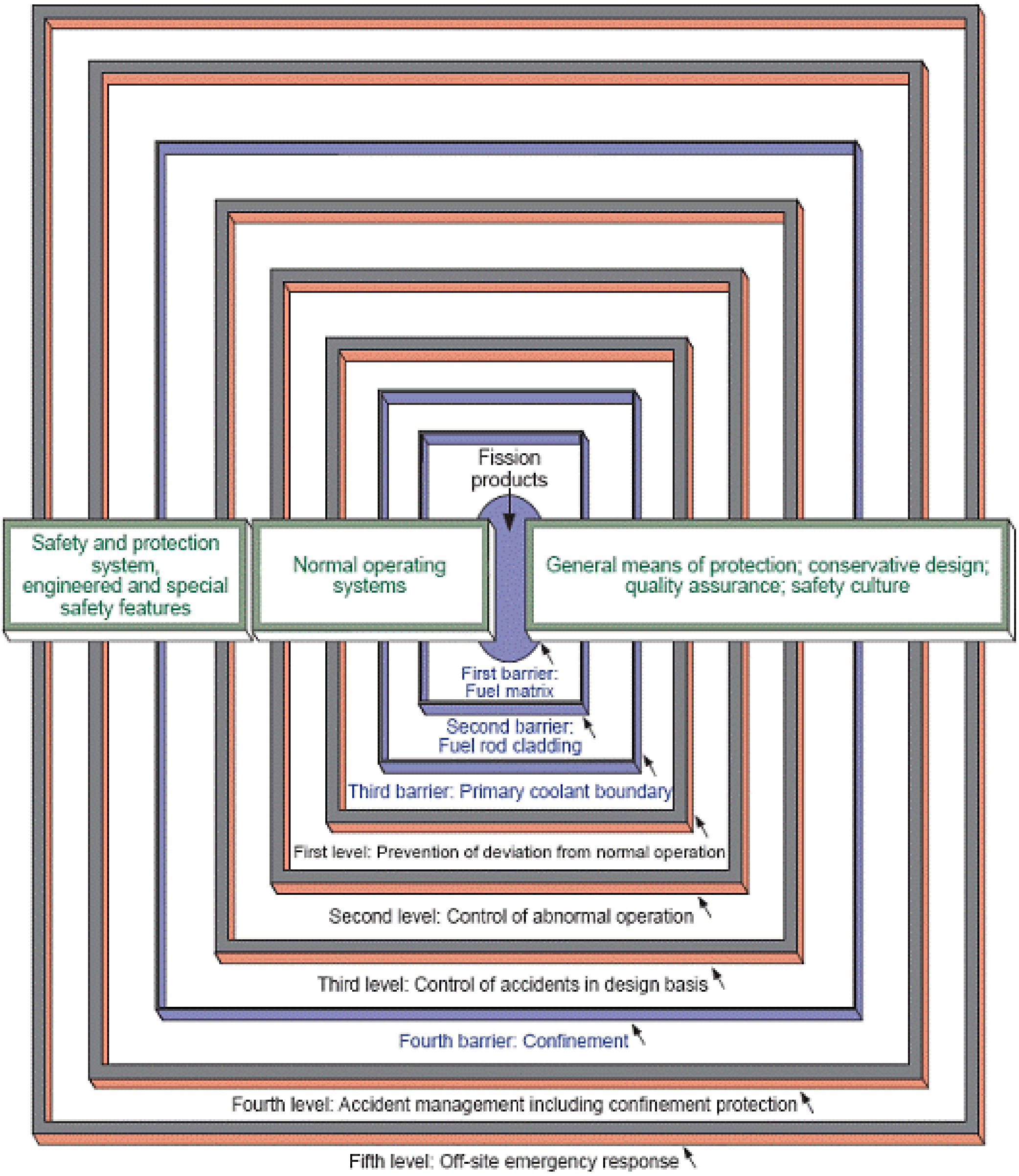


FIG. 4. The relation between physical barriers and levels of protection in defence in depth.

6.1.1.1 Operator's of emergency preparedness organization

Emergency preparedness system is implemented in accordance with the requirements and methodology of IAEA and legislative requirements of the Czech law. Emergency preparedness constitutes one basic part of properties of Czech NPPs. The goal of EP in NPP is to ensure technical, personal and administrative preparedness of plant and external subjects involved in mitigation of extraordinary events consequences with focus to:

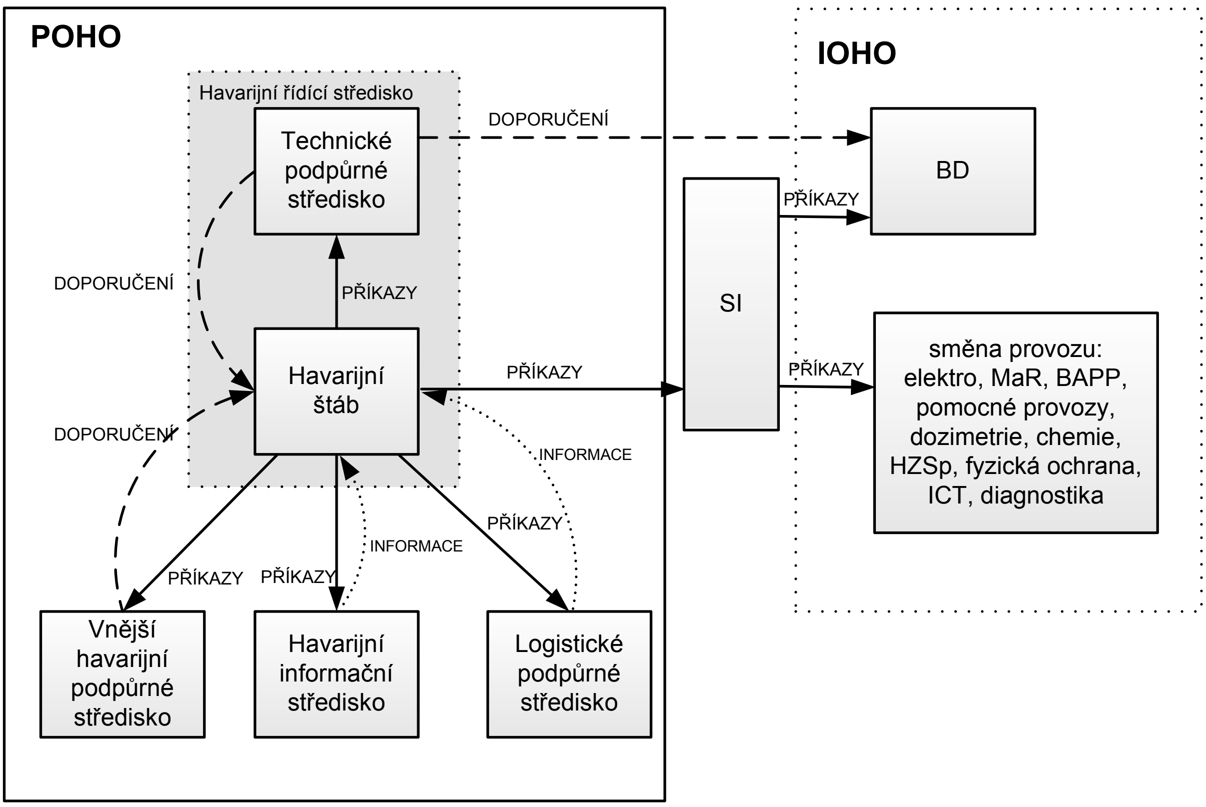
- Decrease the risk of extraordinary event occurrence and/or mitigation of extraordinary events consequences at NPP site and in EPZ.
- Prevention to serious halts deterioration caused by extraordinary event.

Strategic goals of ČEZ, a.s. company are transformed to long term declared goals and tasks in area of EP in accordance with safety politics.

EP strategy is based on logical progress of any event occurring at NPP. For extraordinary events the procedures and instructions are prepared for accident management and performing actions for al staff and other persons included in emergency response organization.

If an extraordinary event occurs in the NPP, the intervention shall be always performed, in the first stage of extraordinary event development, by the continuous shift personnel (IOHO - on-site emergency response organization), under the supervision of the shift supervisor.

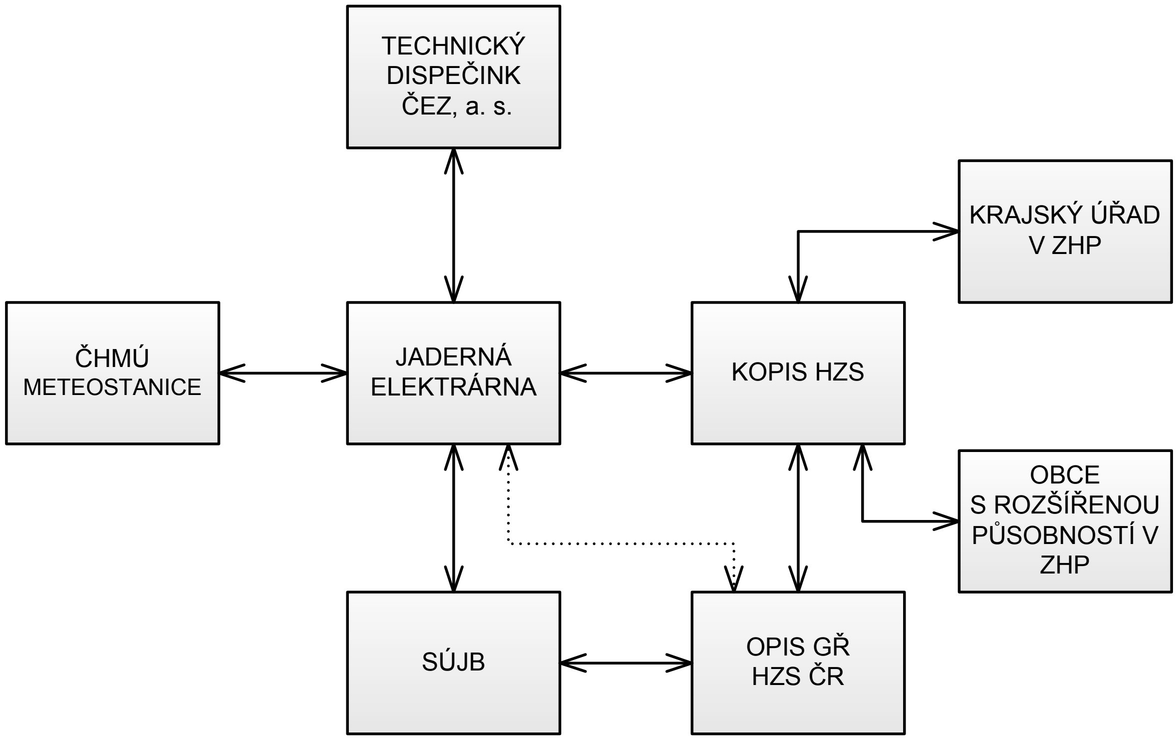
Fig. 6.1.1-2 Emergency response organization with mutual links and information flows



If the scope of the event exceeds the possibilities of the continuous shift personnel, IOHO is complemented with employees on standby within the framework of the emergency response organization (POHO - standby emergency response organization). In that case, the head of the emergency commission takes over the shift supervisor's responsibility when the commission is activated.

If a level 3 extraordinary event occurs, the event shall be announced without undue delay to SONS, the regional authority, the Regional Headquarters of the Fire Brigade, municipalities with extended competences and the Technical Dispatching Centre of ČEZ and the Meteo Station. The information shall be reported by means of the completed form for Initial Report and Follow-up Report on the extraordinary event. Electronic mail or fax is used to send the forms. If it is impossible to establish a direct connection with SONS, a substitute route by means of Operation Information Centre of the Czech fire brigade's shall be used.

Fig. 6.1.1-3 Reporting to authorities in case of level 2 and 3 extraordinary event



For emergency planning to ensure population security in surroundings of NPP in case of radiation emergency and for purpose of off-site emergency planning, emergency planning zone (urgent protective action planning zone) is established and confirmed by SONS (area with radius of 13 km). For measures connected with preparation and execution of evacuation of population, in addition precautionary zone (area with radius of 5 km including municipalities at the border) is established.

Fig. 6.1.1-4 Emergency planning zone

6.1.1.2 Staffing and shift management in normal operation

The staff in the continuous shift operation of the both NPP units (shift staff) works in shifts. The number of the staff members in each shift and the qualifications thereof are sufficient to manage all operating states under normal, abnormal as well as accident operating conditions. The shifts are rotated regularly depending on the shift schedule (shift duration is 8 hours) so that the operating staff has sufficient space and time to relax and to maintain the required qualifications (education, training ...).

The shift operating staff performs all activities prescribed by the operating documentation (procedures, instructions, programs ...) covering normal and abnormal operation as well as accident conditions (including all design basis and partially even beyond design basis events up to fuel damage). Under all of the aforementioned states, the shift staff controls and performs activities with possible support of the other NPP technical staff. In case of the accident conditions occurrence involving fuel damage, the responsibility for the activities control passes to the TSC and EC staff and the shift staff continues performing activities depending on requests issued by TSC and EC.

Fig. 6.1.1-5 Main Control room

The shift supervisor controls the shift under normal operation.

The shift supervisor is responsible for extraordinary events classification, announce and warning of the personnel and activation of required part of emergency response organization. If required he is authorized to activate corresponding part of emergency response organization in advance prior the activation conditions are satisfied. During extraordinary event progress, the shift supervisor is authorized, based on further detail information to perform a change in classification in accordance with current status.

If an extraordinary event occurs, the management of each NPP unit shall be provided for by the following job functions:

- Reactor unit supervisor
- Control room supervisor
- Primary circuit operator
- Secondary circuit operator

The particular main control room is the workplace of these personnel. If it is uninhabitable or there is a loss of possibility to control the unit's technology, the personnel shall perform the activities from the emergency control room.

6.1.1.3 Measures taken to enable optimum intervention by personnel

With a threat to safety on the unit or within the location or in case of occurrence of a situation, which cannot be managed using the shift's own forces, the shift supervisor shall impose one of 3 levels of extraordinary event:

Alert (Extraordinary event of level 1)

Site Emergency (Extraordinary event of level 2)

General Emergency (Extraordinary event of level 3)

To manage extraordinary events, the Emergency Response Organization has been established, which has its internal component (IOHO) consisting of the shift staff and the stand-by component (POHO) consisting of specialists of the NPP technical staff who provide for emergency service (within the framework of 4 shifts). The stand-by POHO component's organization is secured in a manner allowing the respective specialists to reach NPP within 20 minutes during working hours and within 1 hour outside working hours as of the extraordinary event's imposition. The means for the POHO staff's activation are redundant.

Each event significant in terms of safety that could, if not resolved, result in an extraordinary event shall be subject to assessment of deviations from normal operation based on the classification system. The classification of severity of extraordinary events is based on the requirements of SONS Decree no. 318/2002 Coll., as amended, taking into account the IAEA's recommendations stipulated in TECDOC-955 "Generic assessment procedures for determining protective actions during a reactor accident". The purpose of classification of extraordinary events is particularly to ensure timely activation of the emergency response organization and the choice of a suitable and effective response.

The process of assessment of severity of the emerged extraordinary events in nuclear power plants is stipulated in the applicable intervention instructions. The assessment of severity of the reported events shall be performed by a shift supervisor based on a comparison of the type of the reported event with a sum of pre-defined intervention levels. The head of the emergency commission is also obliged to classify extraordinary events. In principle, intervention levels constitute a set of pre-determined, locally specific initiation conditions whose achievement results in assigning the condition of the nuclear power plant to the applicable classification level and type. Intervention levels are prepared for all operating modes of the nuclear power plant. An initiation condition can be an overrun of a pre-determined parameter or the occurrence of specific on-site or off-site events whose spreading could threaten nuclear safety and radiation protection of the nuclear power plant.

<u>Types of extraordinary events</u>

Timely identification of the type of event and the assessment of its severity in terms of safety of the nuclear power plant make it possible to choose an adequate response. From the point of view of their origin, extraordinary events are classified into three basic types:

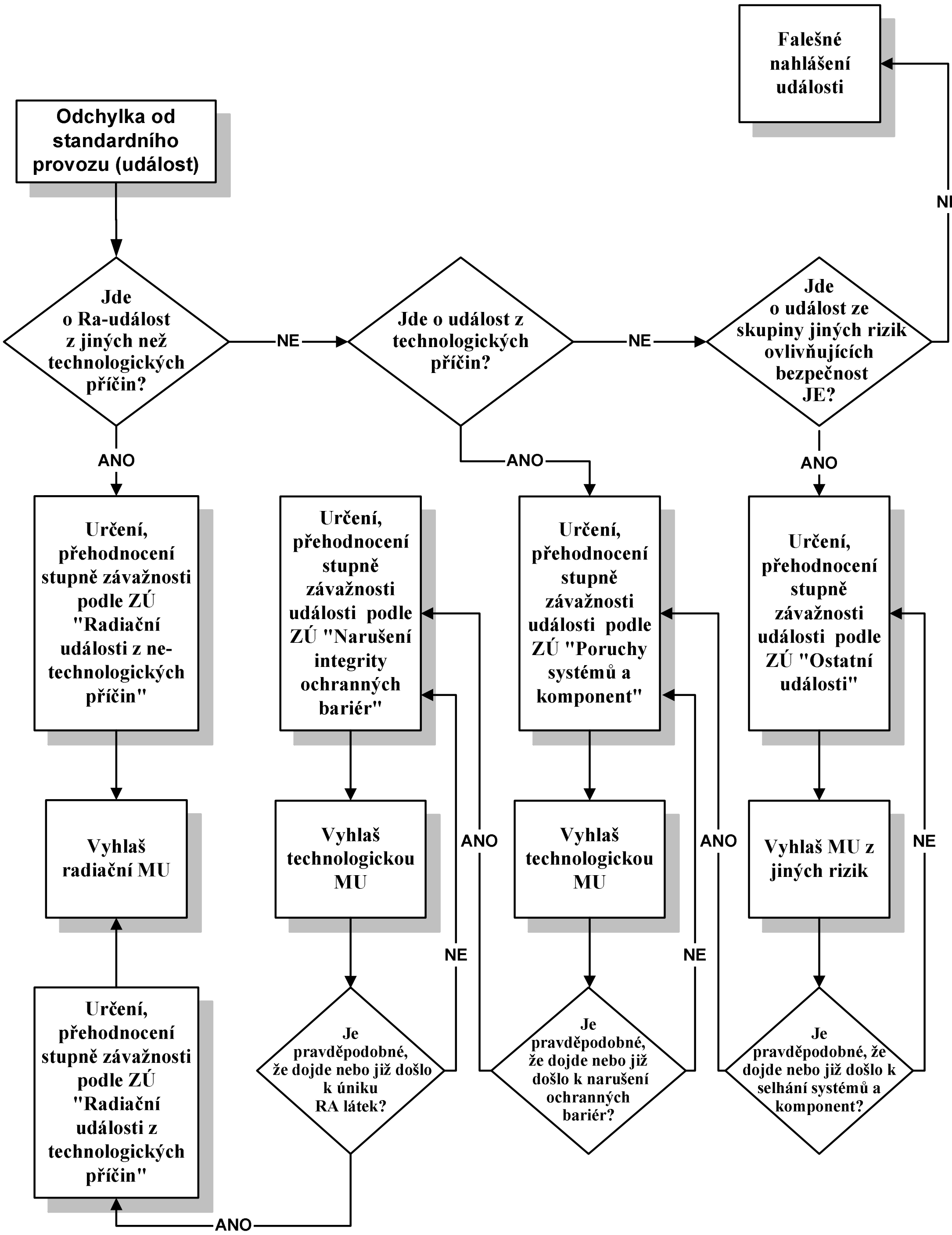
- Events due to technological causes
- Radiation events
- Events due to other risks

This classification of intervention levels allows the shift supervisor to identify the severity of the specific extraordinary event more easily, in particular from the point of view of provision of nuclear safety and radiation protection.

If an extraordinary event is reported, the shift supervisor shall first verify if it is a radiation-related extraordinary event due to non-technological causes. If this possibility is excluded, the shift supervisor shall verify if it is an event due to technological causes and could result in damage to protective barriers and, if applicable, to any subsequent release of radioactive substances, i.e. radiation event due to technological causes.

In case of imposition of Alert, only the Technical Support Centre is activated; in case of imposition of Site Emergency or General Emergency, the Emergency Commission is activated as well. Until its activation, activities are controlled by the shift supervisor and the shift staff proceeds according to the respective operating procedures.

Fig. 6.1.1-6 Fundamental diagram of assessing the type of an extraordinary event



The workplace of TSC and EC is the Emergency Control Centre located within the NPP premises. With imposition of levels Site Emergency or General Emergency, the following entities are activated as well: the so-called Logistical Support Centre (gathering, subsistence and accommodation for the necessary specialists resolving the emergency situation, the Emergency Information Centre (securing public relation with the media and informing the public) and the Off-Site Emergency Response Centre (provision of radiation monitoring in

EPZ) in České Budějovice. All the aforementioned centers are controlled by the Emergency Commission.

The organization-based method of extraordinary events management is defined in the On-Site Emergency Plan approved by SONS.

To resolve technological accidents (up to fuel damage), strategies contained in the Emergency Operating Procedures (EOPs) have been developed. For mitigation of consequences of accidents following fuel damage (severe accidents), strategies contained in the Severe Accident Management Guidelines (SAMG) have been developed. In EOPs, the principal priority shall always be the recovery of heat removal from the reactor core and elimination of breach to the first barrier to prevent leak of fission products (fuel clad), whereas in SAMG, the principal priority is the elimination of breach to the third barrier to prevent leak of fission products (containment), which is the last undamaged barrier at the moment. EOPs and SAMG are symptomatically oriented and are based on the approach adopted from Westinghouse.

<u>On-site emergency response organization</u>

The on-site emergency response organization comprises exclusively shift personnel, i.e. employees who provide for the NPP's normal operation. Based on the shift supervisor's instructions, the continuous shift personnel provides for all activities to mitigate any signs of an emerging extraordinary event until the employees on continuous standby within the emergency response organization are activated.

If an extraordinary event occurs, the shift supervisor shall be responsible for managing the event until his responsibility is handed over to the activated head of the emergency commission. In that case, his activities shall be governed by intervention instructions for shift supervisors, which specify all responsibilities and powers; the most important of them are: assessment of severity of the extraordinary event - classification, information and warning of the NPP personnel and warning within the emergency planning zone, notification to the NPP's management and the competent authorities and organizations, decision to activate POHO, decisions on protective measures for NPP personnel. Responsibility for technology shall rest with the shift supervisor.

If an extraordinary event is declared and depending on its severity, the other continuous shift personnel shall either continue performing their activities in accordance with the applicable intervention instructions and instructions by the shift management personnel or assembly in the operating support centre if protective measures are ordered, from which they shall perform, on the basis of instructions from the shift supervisor or the emergency commission, the required interventions on the technology or provide operating support to the on-line fire rescue brigade during clearing and rescue operations.

To provide for the implementation of protective measures for sheltering and evacuation, shelter groups are appointed to ensure the activation and subsequent operation of shelters within the NPP site. The basic duties of shelter group members in a shelter are: management of the shelter regime, recording sheltered persons, keeping order, operation of air-conditioning and ventilation, radiation monitoring of all persons, DG operation.

Standby emergency response organization

The standby emergency response organization is composed of the personnel of emergency support centers on continuous weekly standby service

- Emergency commission

  The emergency commission is the main managing body of the NPP's emergency response organization. After activation it provides ordering protective measures for the personnel and other persons within the NPP site at the time of occurrence of an extraordinary event, the management of activities of all employees and other persons participating in the intervention to mitigate the development and solve the consequences of an extraordinary event in a nuclear power plant and communication with off-site emergency preparedness bodies. It shall secure the supply of the necessary materials and special equipment, changing of personnel and provision of the necessary materials for them through the logistic support centre.

- Technical support centre

  The personnel of the technical support centre comprise experts from technical departments to provide technical support to control room personnel of affected units during extraordinary event. The technical support centre personnel also provides for an immediate evaluation of the nuclear power plant's safety condition, taking into account nuclear safety and radiation protection, manages the activities of ad-hoc intervention groups to deal with the consequences of extraordinary events and is able to prepare documentation and recommendations for the decision-making and management activities of the emergency commission. If the shift supervisor on duty or the head of the emergency commission requires, it is possible to request support for the technical support centre from other specialists.

Fig. 6.1.1-7 Emergency control centre – emergency commission room

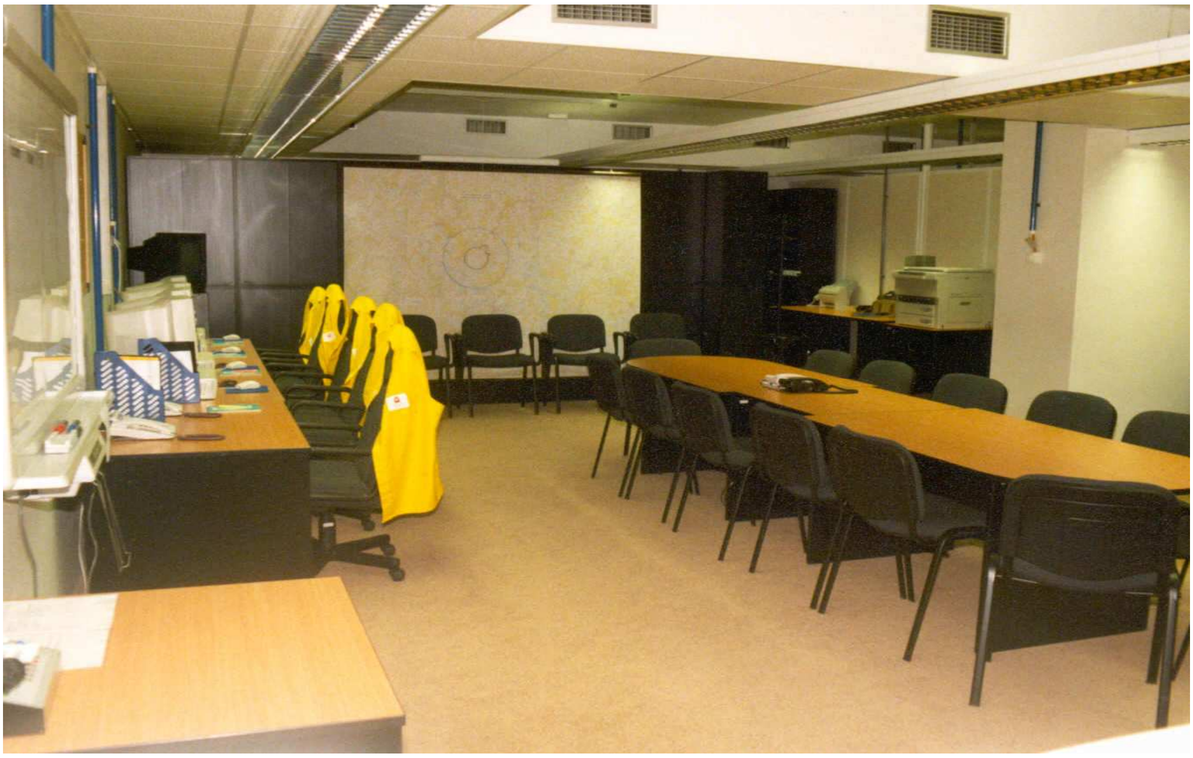


Fig. 6.1.1-8 Emergency control centre – TSC room

- Off-site emergency response centre

  This centre provides for activities associated with radiation monitoring and evaluation of the radiation situation in the emergency planning zone and it shall also prepare projections of further development of the radiation situation in the first stage of a radiation accident based on the radiation monitoring results.

- Emergency information centre

  The emergency information centre personnel shall provide for the transfer of all information to the mass media and for answering all questions the public may have. Their activities particularly focus on informing the general public and public authorities as well as local authorities that are not directly involved in the off-site emergency preparedness system of the nuclear power plant. They are also responsible for preparing press releases for the media. The emergency information centre is located on the premises of the Faculty of Health and Social Affairs, South Bohemian University, Boreckého 27, České Budějovice.

- Logistic support centre

  The personnel of the logistic support centre provides for the necessary technical means and qualified human resources based on the requirements and needs of the emergency commission, the technical support centre and the off-site emergency support centre. The logistic support centre provides off-site support to OHO. The logistic support centre is located on the premises of the Faculty of Health and Social Affairs, South Bohemian University, Boreckého 27, České Budějovice.

6.1.1.4 Use of off-site technical support for accident management

Provision of off-site support and possible use of other capacities, resources and means shall be controlled by EC (Logistic support) in cooperation with the Logistical Support Centre.

Assistance in terms of transport or heavy equipment has been provided on the part of Regional Operations and Information Centre of Southern Bohemia Region, which has the authority to call up other services and organizations to assist in the extraordinary event consequences management. Within the ČEZ Group, assistance is provided by means of the ČEZ Crisis Centre for the affected location. Within the framework of this body, availability of external specialists (suppliers, specialist knowledge base, international assistance, etc.) would be provided in case of necessity. The most efficient assistance is assumed from the Dukovany NPP location.

Many authorities and organizations at the national as well as local level are involved in the provision of off-site emergency preparedness. In case of an extraordinary event the nuclear power plant communicates with the following off-site authorities and organizations on the both national and local level.

- SONS Crisis Centre

  SONS Crisis Centre provides, through the radiation monitoring network of the Czech Republic, for an independent evaluation of radiation signs due to a radiation-related extraordinary event. Based on the monitoring results of the individual elements of the Czech Republic's monitoring network, it provides documents for decisions of the Regional Crisis Centre on population protection measures.

- Regional authority

  Regional authority provides for the coordination of off-site emergency preparedness of all municipalities with extended competences whose territory overlaps with the emergency planning zone. In cooperation with the mayors of the affected municipalities with extended competences, the governor of the respective administrative region deals with all activities associated with the provision of off-site emergency preparedness within the whole emergency planning zone and makes decisions on announcing and implementing population protection measures. His advisory board is the Regional Crisis Centre. Emergency protection measures are declared on the basis of recommendations from the SONS Crisis Centre, prepared on the basis of radiation monitoring results and other input provided by the individual elements of the radiation monitoring network.

  In case of a radiation accident in a nuclear power plant, the operator provides the necessary cooperation, data and information essential for assessing the severity of the situation through the regional emergency commission. To ensure cooperation, the nuclear power plant shall send its representative to the Regional Crisis Centre.

- Municipalities with extended competences

  The mayors of municipalities with extended competences shall decide on summoning the municipal crisis centers and shall be in charge of declaring and implementing protective measures on the concerned territory of the municipality. They shall manage these activities in accordance with the Off-site Emergency Plan. The protective measures shall be declared subject to prior discussion with the regional crisis centre, which provides for the mutual coordination of reports and information passed between the individual municipalities with extended competences, SONS and the nuclear power plant. This procedure is used to ensure consistency of the declared protective measures on the territories of the individual municipalities with extended competences.

- The fire rescue brigade

  The fire rescue brigade provides, on the basis of an instruction from the nuclear power plant, for a warning to the population living in the emergency planning zone by means of hooters controlled by the national integrated warning system and makes sure that the needed TV and radio announcements are broadcast. The regional fire rescue brigade shall also make sure for ČEZ, a.s. that the concerned municipalities with extended competences are informed through regional operation and information centers of the fire rescue brigade (in compliance with Decree No. 318/2002 Coll., as amended).

- The Czech Hydrometeorologic Institute

  The Czech Hydrometeorologic Institute shall provide for an evaluation of the current meteorological situation and preparation of projections of further development for the nuclear power plants. The outputs from basic meteorological data necessary for assessing the potential or actual dissemination of radioactive releases in the vicinity of the nuclear power plant shall be sent to the specific information networks of the power plant.

- State police and security guard

  State police and security guard cooperate during warning to the population in EPZ, evacuation organization, traffic control, guarding of buildings etc.

- Medical rescue service (Traumatological plan)

  At NPP site there is a permanent medical rescue service. During normal working time there is a physician and nurse present ate the site. Permanent service is ensured by emergency physician, nurse and ambulance car driver.

Fig. 6.1.1-9 Off-site emergency preparedness

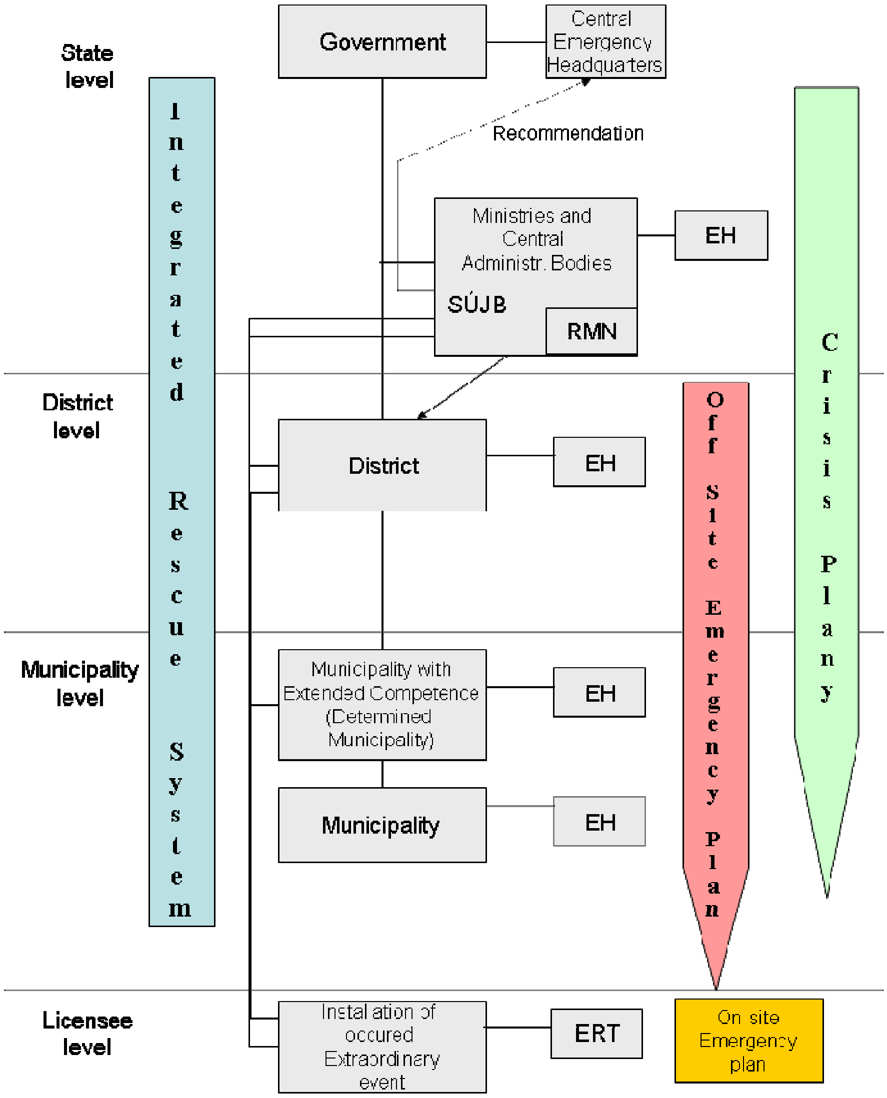
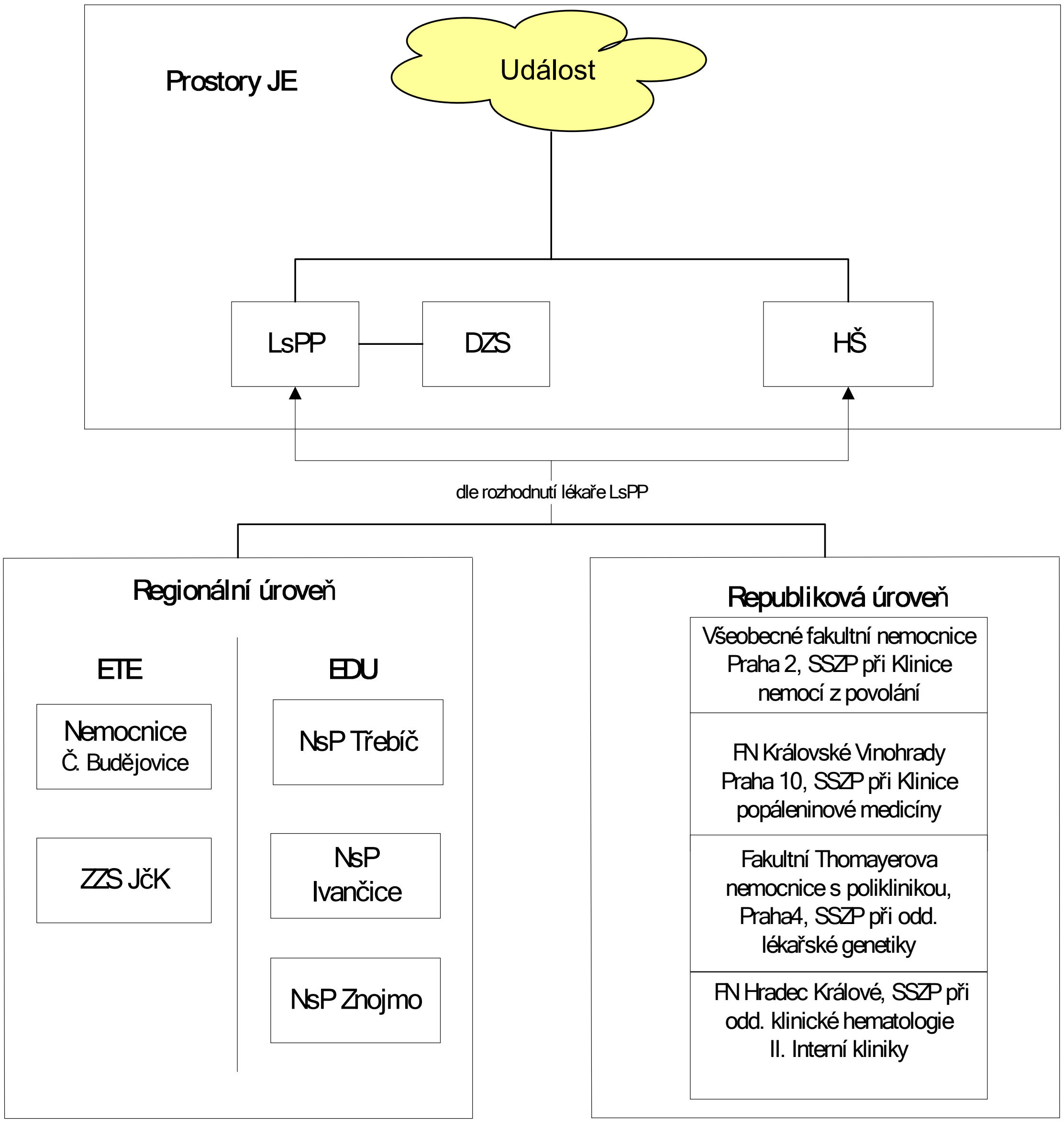


Fig. 6.1.1-10 Relationships with external medical institutions and provisions



6.1.1.5 Procedures, training and exercises

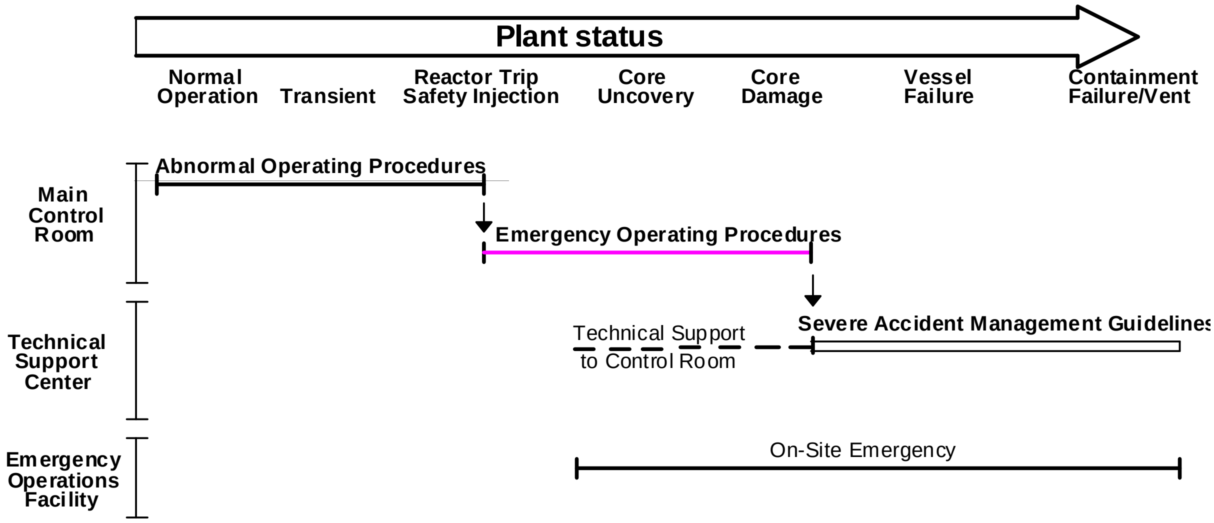
Accident management program is based on symptom oriented approach. At the present time, the following strategies to manage design basis, beyond design basis and severe accidents are prepared for NPP:

- Symptom based Emergency Operating Procedures for power states (EOPs)
- Symptom based Emergency Operating Procedures for shutdown states, including cases of threat to heat removal from spent fuel deposited in SFP (SEOPs)
- Manuals for decision-taking by TSC

* Severe Accidents Management Guidelines for power states (SAMG)

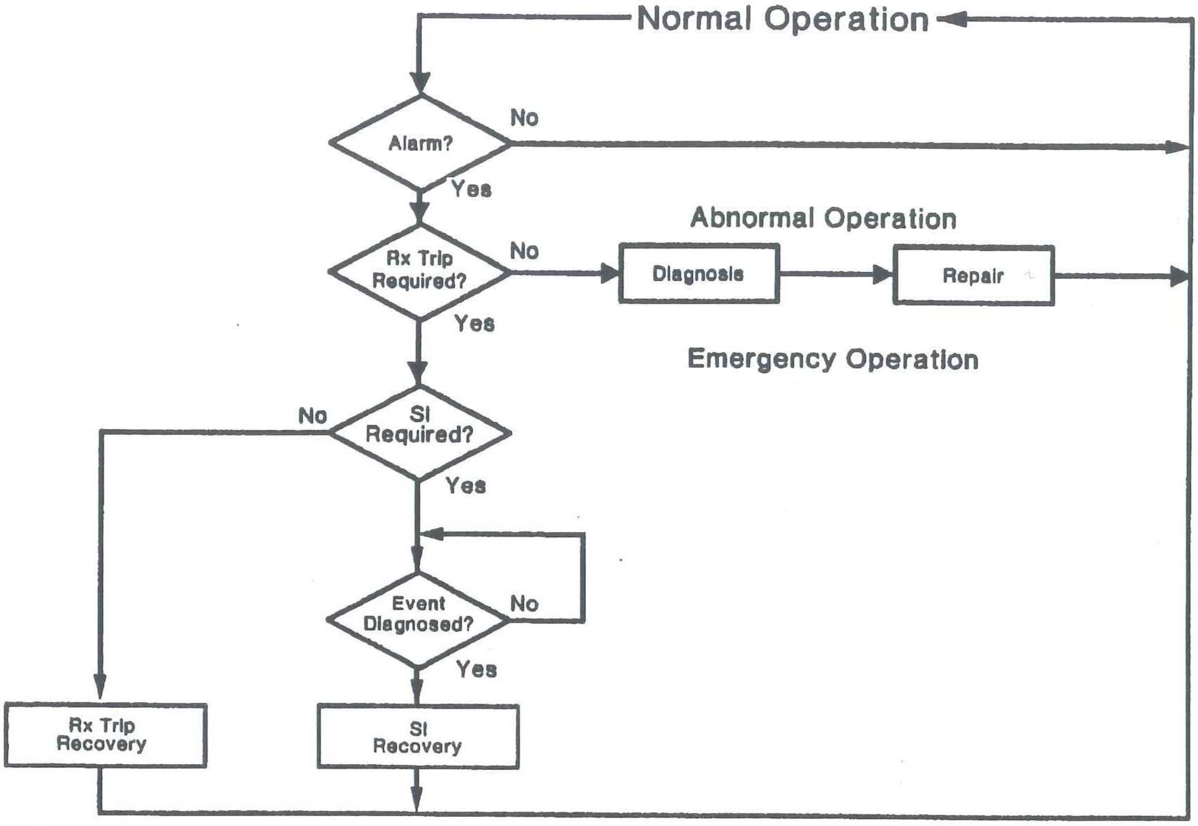
In addition to the type of use of operating documentation during activities responding to given situation, the progress of emergency situation development is also closely related to activities of emergency response organization in accordance with the On-site Emergency Plan (declaration of the extraordinary event level).

Fig. 6.1.1-11 Relationship between operating conditions and extraordinary events



Activities of the operational personnel at each level are controlled by the operating procedures, adopted for each operating status. The procedures form a network which identifies activities of operating personnel at each operating status of the unit.

Fig. 6.1.1-12 Control room personnel response



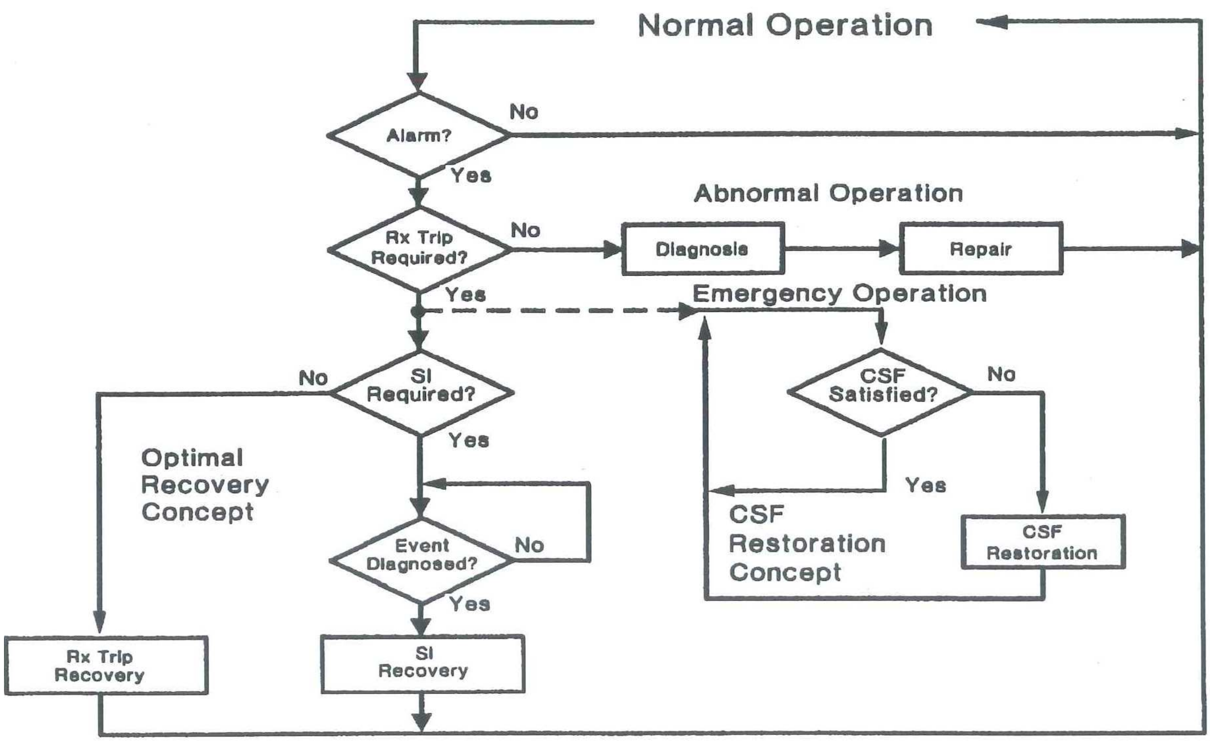
Strategies contained in the Emergency Operating procedures (EOPs) have been developed for emergency conditions in the preventive phase. Strategies contained in the Severe Accident Management Guidelines (SAMG) have been developed for severe accidents management. The basic precondition for performance of activities in accordance with the emergency procedures is achievement of such status of the core, which allows its cooling, i.e. core in coolable geometric configuration. In case of irreversible damage of the core the emergency procedures do not need to provide optimum instructions for dealing with the emergency situation and it is necessary to initiate activities in accordance with SAMG. Also the main priority change at this moment. In EOPs, the principal priority shall always be the recovery of heat removal from the core and elimination of damage to the primary barrier to prevent leak of fission products (fuel clad), whereas in SAMG, the principal priority is the elimination of damage to the tertiary barrier to prevent leak of fission products (containment), which is the last undamaged barrier at the moment.

The purpose of the actions described within EOPs, which are to be used by the operating personnel for dealing with design-basis and beyond design-basis accidents, is to provide sufficient cooling of the core and so prevent irreversible damage of the core and further minimize consequences of any possible leakage of radioactive material out of the power plant. Philosophy of these procedures includes permanent evaluation of the status of physical barriers preventing leakage of radioactivity by assessment of critical safety functions. This evaluation provides early identification of deterioration of safety status of the unit and also provides possibility of early correction in case negative trend of event development is identified.

The set of symptomatically oriented emergency operating procedures provides a systematic tool to the operating personnel (independent on the course of the emergency mode) for dealing with emergency situations, by means of a set of predefined and structured emergency procedures. Combination of event and function oriented strategies provides instructions to the operating personnel for bringing the unit to safe end state, while providing permanent unit condition diagnostics and possible recovery of safe status independently on the course of the given emergency situation.

In addition the emergency procedures contain systematic tools for evaluation of safe status of the unit by assessment of status of critical safety functions. The critical safety functions are closely related to physical barriers which prevent radioactivity releases to environment.

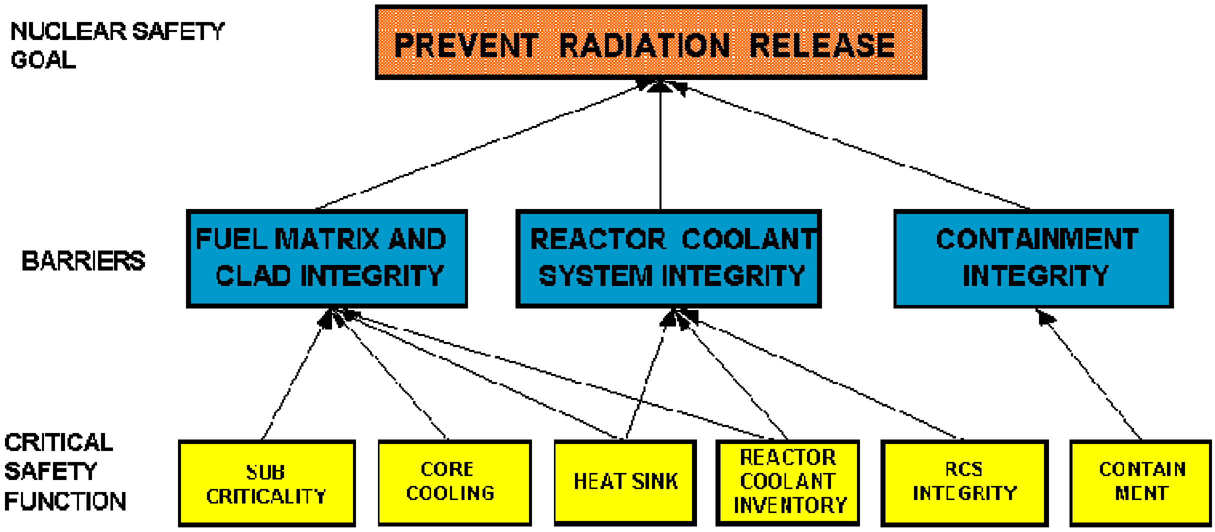
Fig. 6.1.1-13 Plant control during emergency conditions



A set of safety functions exists for each of the above physical barriers, which must be fulfilled to ensure their integrity. If this set of safety functions is fulfilled full protection of population against possible consequences of nuclear power station operation is provided.

Support from TPS personnel is provided for the preventive phase of emergency situation management, when operating personnel proceeds in accordance with EOPs. For this purpose TPS personnel is provided with „TSC evaluation manuals“, which provide basis for decision making in supporting the operating personnel during their activities in accordance with the emergency procedures.

Fig. 6.1.1-14 Relationship between safety functions



These instructions were developed for TSC personnel and for other technical personnel of NPP, which is authorized to provide decision making support in addition to TSC personnel. Among these personnel belongs Shift supervisor or Unit supervisor, who can find number of qualified decisions in this document, concerning further activities in accordance with EOPs.

- The TSC manuals are used by TSC personnel, in case TSC is functional (declared Alert, convened TSC and TSC personnel is capable to provide support).
- Instructions in these Manuals are used by Shift supervisor or Unit supervisor, if decision making support is required before TSC becomes functional.

TSC personnel provide support by evaluating current situation in accordance with the instructions and by provision of recommendations for use of emergency procedures.

In case an event develops to a severe accident, further procedure is chosen with regard to provision of at least the remaining barriers against radioactivity leakage. Under such conditions it is not possible to apply EOPs anymore. SAMGs are available for this phase which control activities leading to achievement of controlled stable state.

Transition to SAMG takes place in case irreversible damage of the core is identified. In such a case activities according to EOPs are terminated and transition to SAMGs takes place. The only entry point to SAMG is the instruction SACRG-1, SEVERE ACCIDENTS CONTROL ROOM GUIDELINE INITIAL RESPONSE.

There are three possible transitions from EOPs to SAMG:

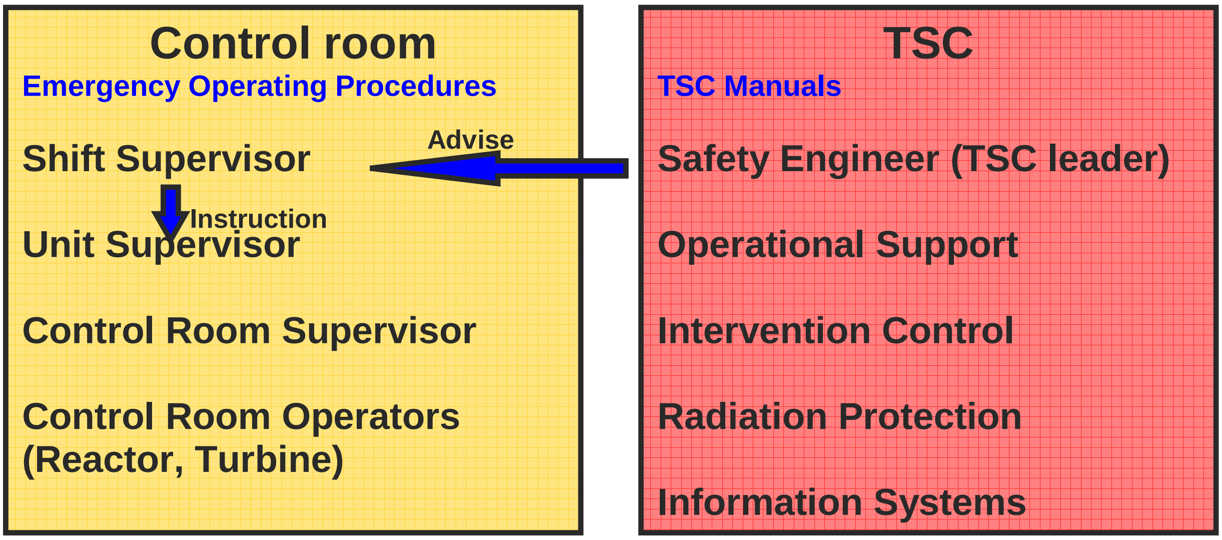
FR-C.1 Response to inadequate core cooling

FR-S.1 Response to nuclear power generation - ATWS

ECA-0.0 Loss of all safety AC 6 kV power

These three options of transition from emergency procedures to SAMG are sufficient and cover all possible scenarios of severe accidents.

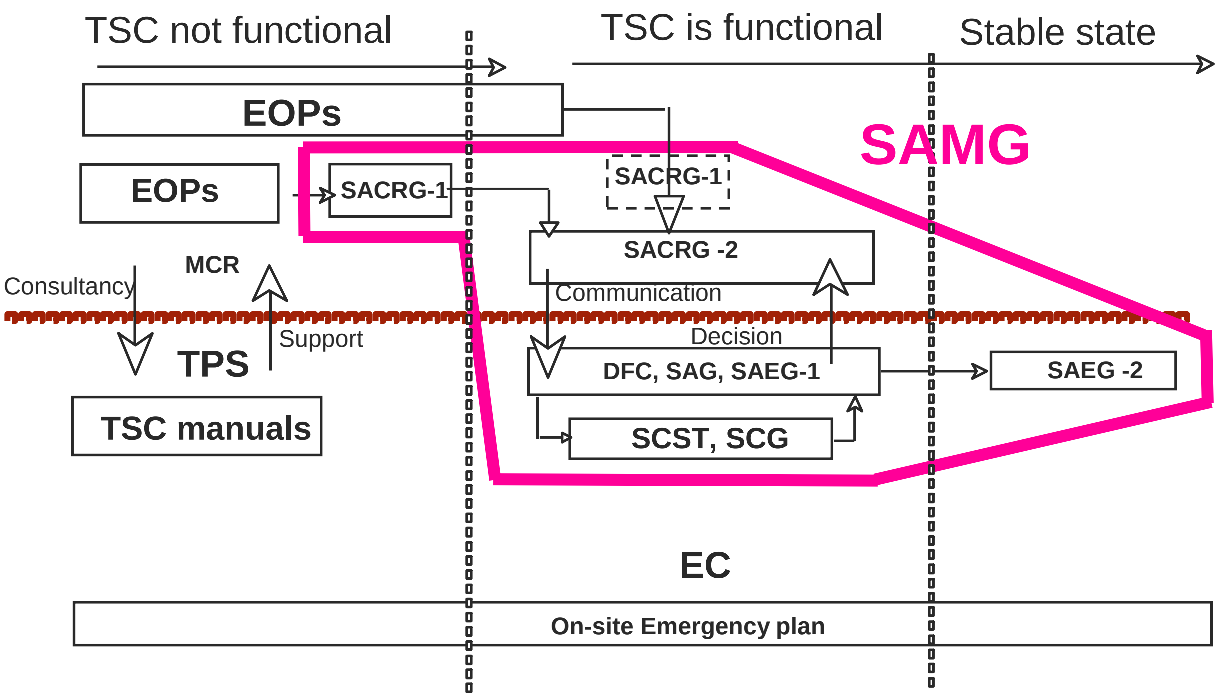
Fig. 6.1.1-15 Communication between TSC and control room while using EOPs



The following SAMG objectives need to be achieved to mitigate consequences of severe accidents.

- Recovery of heat removal from the core or the melt = recovery of controlled and stable state of the heat generation source
- Maintain integrity of the containment as the last barrier against radioactive materials releases to environment = provide controlled state of the containment
- Terminate releases of radioactive materials to environment
- Provide maximum operability of systems for fulfillment of primary objectives

Fig. 6.1.1-16 EOPs and SAMG relationship

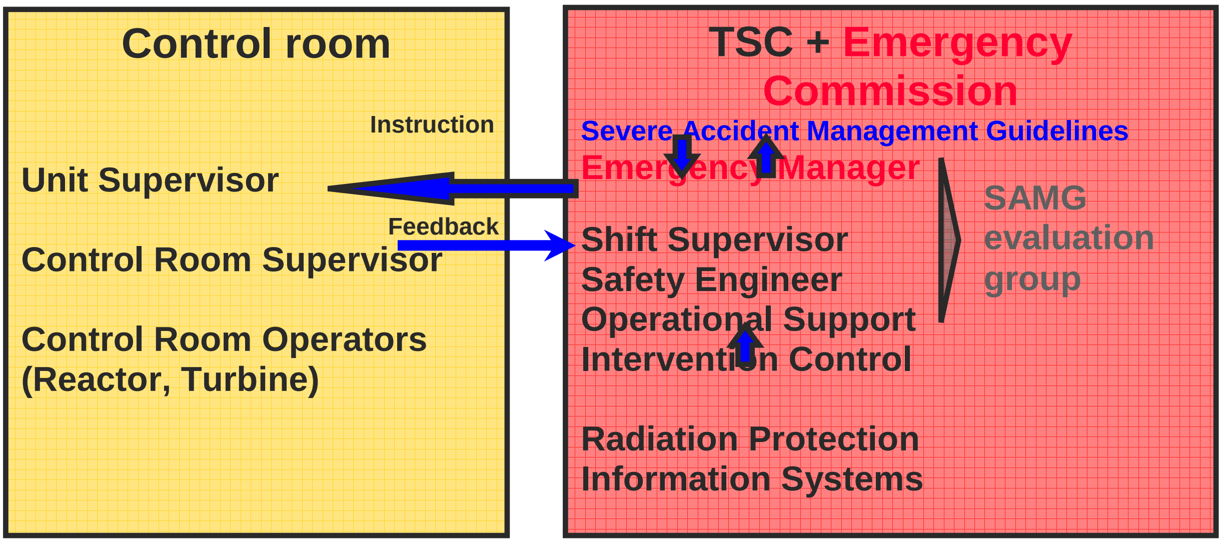


The main content of SAMG strategies is based on systematic approach to evaluation of actual state and identification of actions solving the given state.

- Diagnostics of the unit's state using measurable and available parameters;
- Determination of priorities for further solution and selection of possible interventions to solve the specific state to minimize its consequences;
- Evaluation of availability of equipment necessary for the recommended interventions and, if applicable, determination of priorities for requirements on the recovery of the necessary equipment;
- Identification and evaluation of possible negative impacts of the recommended interventions, determination of their acceptability and identification of possible remedial actions to mitigate the effects in case of negative impacts;
- Recommendations for performing a specific intervention on the basis of comparison of the possible further development without an intervention vs. development in case of intervention and the occurrence of accompanying negative impacts;
- Determination of the final effectiveness of the performed interventions and recommendations for performing remedial actions in case of negative impacts;
- Identification of additional activities to secure long-term state of the unit.

In case of termination of activities according to emergency procedures and transition to SAMG also activities in accordance with TPS instructions are terminated and further activities are controlled by SAMG.

Fig. 6.1.1-17 Communication between TSC and control room while using SAMG



The entire process of EOPs and SAMG implementation is based on assumption of a symptomatic approach to plant control in emergency situations, developed within Westinghouse Owners Group for Westinghouse units in USA and elsewhere in the world and its application to VVER design. In addition the proved approach to verification, validation, implementation and training has been assumed.

For the maintenance of EOPs, SEOPs and SAMG, updating thereof is performed on a regular basis consisting of knowledge based on training on a full scope simulator and knowledge acquired during exercises. External knowledge (within the framework of the "users group") is reflected in the documentation in the "Maintenance program".

For the selection of the shift employees and for the selection of the POHO employees, a system based on qualifications requirements has been implemented and other criteria taking into account such employees´ knowledge and specialization are considered as well. The forms of preparation of the NPP staff, by means of which the necessary qualifications are secured, including the control room staff and the TSC support staff to manage extraordinary events on NPP, are as follows:

- Basic preparation / re-qualification
- Periodic preparation (including emergency preparedness training)
- Training on a full scope simulator (including the MCR and TSC staff's cooperation)

- Training for the selected staff concerning the issue of beyond design basis and severe accidents

For the respective positions of the shift and support staff, staffed to provide for the performance of activities concerning accidents management, the personal and qualifications-related requirements have been defined, and these requirements are inspected by means of a set of qualifications requirements as well. For every position, requirements concerning education, specific knowledge (basic preparation, periodic preparation, training on a simulator ...) are prescribed, including the career-development training. For the selection of the POHO employees, a system based on qualifications requirements has been implemented and other criteria taking into account such employees´ knowledge and specialization are considered as well. It is necessary to adopt additional measures in particular to improve selection of personnel with the best knowledge in the area of severe accident management.

Preparedness of the shift and the technical staff to manage technological accidents, is tested during training on a full-scope simulator in the presence of the TSC staff and in the course of the emergency exercises. The emergency exercises are conducted at least 4 times a year so that each shift completes the POHO exercise at least once a year. The exercises include the preparation for the variants of operational interventions under aggravated conditions. Relevant procedures have been prepared for activity of the intervention teams under aggravated conditions and for safety thereof. A real training (drill) concerning the use of SAMG for severe accidents management on NPP was performed after the SAMG implementation.

6.1.1.6 Dependence on the functions of other reactors on the same site

Both NPP units are mutually independent in terms of technology and separated in terms of construction design. The joint systems common to both units are raw water supply from the Vltava River and ECW cooling pool for heat transfer from reactor core, SFP and safety systems into the atmosphere as the ultimate heat sink.

In case of loss of raw water supply to NPP, each of the three ECW cooling pools is able to remove all heat from both units for 12.5 days, without any refilling. The analysis of usability of fire-fighting equipment on the site implies that it is possible to refill the inventory in the operated ECW cooling pool by pumping with the use of mobile equipment from the remaining ECW cooling pool.

In addition to ECW cooling pool (passive, seismically resistant structures), all other technological systems for heat transport are mutually independent and separated in terms of construction design for both units.

Taking into account the independency of power supply for both units from off-site and on-site power sources (incl. emergency sources), it is possible to use the power supply sources of the adjacent unit advantageously in case that an SBO occurs on a unit.

Another joint system that can be relevant for severe accident management is the inventory of boric acid solution, stored for both units in the auxiliary building. This comprises a sufficient inventory of 1,600 m^3, which is available for both units (at a volume comparable to the volume of boric acid solution available in the containment sump).

A SFP is located in the respective containment for each unit. This layout is advantageous to prevent any leakage of fission products in case of damage to the radiated fuel stored in SFP. A disadvantage of this layout is a difficult access to SFP for the purpose of emergency make-up feeding with other emergency means (fire fighting equipment etc.). Likewise, the SFP can be also affected in case of an accident to the reactor's equipment placed within the containment, and vice versa.

6.1.2 Possibility to use existing equipment

6.1.2.1 Provisions to use mobile devices (availability of such devices, time to bring them on site and put them in operation)

To provide for safety functions (concerning the design basis as well as beyond design basis scenarios), the respective symptom based procedures and guidelines have been prepared (EOPs or SAMG). No mobile or non-technological means from off-site sources are assumed in EOPs or SAMG (with the exception of means of the On-site Fire Rescue Brigade).

Fire brigade consisting of 16 firemen per shift (non-stop 24 hours service) is available at Temelin NPP. The brigade operates appropriate fire fighting technology and is trained to manage any fire at any place of the location.

The HZSp pumping technical equipment counts among the main mobile non-technological means usable for transport and media supply. It is designed to pump-out water during possible floods. For pumping and transport of water the fire brigade has available 4 tank-truck fire engines, 1 combined fire truck and 3 trailer fire-engines, with the total nominal capacity of 280l/s.

On-site Fire Rescue Brigade equipment has not been supposed yet to support mitigative actions in case of technological accident. Hence, the capability of this equipment for technological purposes is not verify and no connections to plant systems exists to provide fundamental safety functions.

The fire-suppression equipment and the intervention staff are located in the civil structure of the Fire Station where no direct effects of extreme natural hazards can be expected. However, due to extreme natural hazards the Fire Brigade intervention might be limited. In case of a seismic event a drive out of fire fighting technology to open areas is assumed immediately at occurrence of first symptoms of earthquake, in order to prevent any damage by falling structures and plant.

6.1.2.2 Provisions for and management of supplies (fuel for diesel generators, water, etc.)

Fuel oil management is available on the, used as an inventory for the long-term operation of DGs; it can be also used for additional mobile Diesel generating sets, if necessary. All designed Diesel generating sets available on the site have their own fuel oil tanks, dimensioned to ensure autonomous operation (without fuel oil refill) at the maximum load:

- Safety DGs for at least 48 hours (for a longer period of time in reality)
- Common DGs (supply for consumers on both units) for about 12 hours

If long-term operation is needed, it is possible to bring additional fuel oil using a connection pipe stretching along technology bridges from fuel oil management. Fuel oil management pumps are not supplied from secured power supply systems, and further fuel oil refill would be therefore ensured using tank trucks.

All auxiliary systems of the motor and DG (fuel supply to the motor, lubricating oil, internal cooling circuit, boost air, actuation air) are autonomous and, during DG operation, independent of off-site power supply. Systems that could be affected by a long-term operation of DG (e.g. clogged lubricating oil filters) are redundant subsystems whose one section can be shut down during DG operation, conduct the necessary maintenance, and thus prevent a failure of DG due to a loss of auxiliary systems. The quality of fuel oil is inspected regularly once a month and is maintained in accordance with the applicable requirements.

When demineralized water is refilled to SG, storage tanks 3 x 500 m^3 of the emergency SG feed-water system are available for each unit, plus the tanks 2 x 770 m^3, common to both units. This water inventory easily suffices for unit cool-down into a cold shutdown state (the design basis specifies one emergency SG feed-water system as sufficient for cooling down the unit into a cold shutdown state) or for maintaining the units in a hot state for approximately 72 hours.

Each ECW cooling pool is able to remove all heat from the unit with a shut-down reactor without make-up feeding in a long-term setting even in the most unfavorable condition, without the ECW temperature significantly exceeding the maximum design-basis value. The most unfavorable case is a failure with a LOCA on one unit and shutdown of the other unit, i.e. the heat source for ECW is maximal. Thanks to the existence of three redundant ECW systems it is possible to establish that heat removal to the ultimate heat sink can be secured at least for 30 days on condition that all safety divisions are gradually used or that the inventory of water from ECW cooling pool of operable ECW systems is pumped using mobile devices to ECW cooling pool of an operable ECW system.

6.1.2.3 Management of radioactive releases, provisions to limit them

The purposes of all the strategies to manage accidents (in EOPs as well as SAMG) aim at preventing leaks of radioactive media to the environment and thus eliminating damage to health and safety of persons. If, in spite of the aforementioned measures, leak of radioactive substances occurred in the course of the accident development, then all activities pertaining to accident management shall aim at terminating or at least limiting such leaks.

To create forecasts of consequences of a possible leak of radioactive media and for the purpose of assessment of the current radiation situation in case of leak, NPP uses the RTARC software utilizing the current meteorological data, the meteorological data forecast, data concerning leak, the plant buildings dimensions, data concerning the land relief of the NPP environment and data concerning the leaking radionuclides. The output of the program is the current radiation situation and its forecasts for the selected period in the NPP environment from 500 m to 40 km.

Restricting the exposure of persons and the environment during a radiation extraordinary event is performed by means of protecting measures, which are as follows:

- The urgent measures including sheltering, iodine prophylaxis and evacuation.
- The follow-up safety measures including changing residence, control of food and water contaminated by radionuclides and control of animal fodder contaminated by radionuclides.

The protecting measures during radiation accidents shall always be taken if substantiated by a higher benefit than expenses incurred in relation to measures and damage caused thereby, and they shall be optimized as far as their form, scope and duration are concerned so that they result in the highest reasonably realizable benefit.

Depending on the level of the radiation situation, the respective urgent safety measures would be imposed in case of imposition of an extraordinary event so that it is secured that the intervention levels defined in the following table are not exceeded; the table contains the radiation exposure limits for the staff members and other persons for the imposition of safety measures in case of the extraordinary events´ occurrence relating to the time of 8 hours as of the extraordinary event's occurrence.

The aforementioned limits do not apply to the radiation exposure of persons participating in interventions in case of a radiation incident or accident, but this exposure shall not exceed ten times the basic limits for workers with sources (the basic limit is 50 mSv per calendar year, or 100 mSv for five consecutive calendar years), unless this concerns lifesaving efforts or efforts to prevent the development of a radiation accident, with possible extensive social and economic consequences. The employees who would perform the intervention are informed of the risks and the expected exposure dose prior to the intervention.

Tab. 6.1.2-1 Protecting measures for personnel

	PROTECTING MEASURES		
CATEGORY OF PERSONS	**Sheltering**	**Iodine prophylaxis**	**Evacuation**
Other persons and employees not included in OHO	5 mSv	5 mSv	5 mSv
OHO staff	50 mSv	5 mSv	200 mSv
The OHO staff in case of rescuing lives or preventing the extraordinary radiation situation development	Pursuant to sect. 4(7) c of Act No. 18/1997 Coll.	5 mSv	Pursuant to sect. 4(7) c of Act No. 18/1997 Coll.

Four shelters have been constructed at NPP. In one of the shelters, Emergency Control Centre has been built, in another one, the Operational Support Centre where all the shift staff members necessary for the execution of local interventions would be gathered. The capacity of the shelters in the plant is 1,755 persons to be sheltered. Each staff member in the workplace or in the shelter shall have personal protective equipment at their disposal to protect against surface and interior contamination.

Iodine prophylaxis using the potassium-iodine tablets is performed in all NPP employees and persons living in EPZ who are personally equipped with potassium-iodine tablets. The persons residing within EPZ have been equipped with a handbook for the protection of residents containing instructions in case of radiation accident. The handbook contains instructions concerning warning by means of hooters and instructions concerning initial safety measures (sheltering, iodine prophylaxis and evacuation).

6.1.2.4 Communication and information systems (internal and external)

The overview of the internal communication means:

- The system of warning and informing the staff members (hooters in civil structures, internal hooters, plant and operation internal radio)
- Telephone central
- Telecommunication dispatching equipment
- Radio networks
- Local utility notification system (PAGING)
- Radio networks amplifiers (radiation cables)
- Manual radio stations
- Mobile radio stations
- Communication system to summon POHO and the Alarm input terminal

The overview of the external communication means:

- The system of warning and inform the residents (hooters in EPZ)
- Pre-prepared recordings for the state mass media (TV, radio)
- The O2 operator telecommunication network (the mobile as well as the fixed one)

For all the communication systems, the method of confirmation and verification based on periodic checks has been set.

- Once in three months for the functionality of technical equipment, systems and methods of activation of persons for intervention management and performance.
- Once in six months for the functionality of technical equipment, systems and methods of warning of the personnel and other persons within the NPP site.
- Once in three months for the functionality of technical equipment, systems and methods of reporting extraordinary events and notification of radiation accidents.

- Twelve times a year for the functionality of technical equipment, systems and methods for warning the population in the emergency planning zone.

The records on the performance and results of checks of functionality of technical equipment, systems and methods for information and warning are kept in the archives and shall be stored there for three years.

To activate the members of the Stand-by Emergency Response Organization and to activate the hooters, there are minimum two ways available (autonomous and independent of the possible overload of the mobile telecommunication networks).

The Emergency Control Centre is equipped with the information system providing for access to all information necessary for the extraordinary events management. In case of occurrence of an extraordinary event, there is a sound contact from TPS to the both CRs. The both MCR and TSC is equipped by closed-circuit television system capable to monitor all essential parts of the plant. The TSC staff has the current on-line technological as well as radiation data at their disposal, with which the staff performing operational management works as well.

6.1.3 Evaluation of factors that may impede accident management and respective contingencies

6.1.3.1 Extensive destruction of infrastructure or flooding around the installation that hinders access to the site

Access to the essential civil structures could be limited due to destruction of structures not having sufficient seismic resistance onto the on-site access communications, as well as due to a fall of debris into the area of the plant entrance. In such a case, it could be possible to use the stand-by access communication/entrance to the plant premises.

6.1.3.2 Loss of communication facilities / systems

The internal radio system is without stand-by power supply. The stand-by power supply for the operation of the communication means for the purpose of both warning within the location and communication of the key staff members (Emergency Control Centre, shelters, HZSp, SONS, IZS, CR staff) is secured in the order of hours in case of loss of power supply or in case of damage to the infrastructure. The hooters in the civil structures have own accumulator batteries.

In case of a long-term SBO, a loss of power supply of the telephone central and telephone centrals of the off-site cooperating network workplaces might occur, with the exception of transmission grid operator centers, which are equipped with own DGs. In this manner, recovery of power supply from external power supply sources is threatened.

Recovery of power supply from power supply sources outside NPP (e.g. from ELI) is conditioned by cooperation (necessary connection) of several entities (ČEZ, ČEPS, E.ON).

The fixed telephone network, the mobile network, the wireless stations, the means of warning, etc. are not secured against global flooding. Communication would be possible via the HZSp wireless stations.

In case of damage to the infrastructure, communication between the persons performing intervention and the control centers might be threatened as well as with the off-site centers of the public administration bodies (SONS Coordinating Crises Center, the Regional Crisis Centre, IZS, etc.).

6.1.3.3 Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities on site

In case of damage of NPP or infrastructure the Emergency Control Centre or shelters could be threaten or lost. In case it is not possible to use the Emergency Control Centre for any reason, the stand-by centre located in České Budějovice (outside of EPZ) which disposes of limited amount of information to manage extraordinary situations. In case of inaccessibility of the shelters the situation could be solved by evacuation of not needed personnel outside of the plant. Inaccessibility of the Emergency Control Centre could be solved by managing of the actions from MCR or ECR. Crisis plans with the use of support have not been prepared.

In case of inaccessibility of the plant, the situation would be resolved by restricted rotation of the staff members, their accommodation directly in the location or nearby (in shelters and Emergency Control Centre, the possibility to use the information centre building).

In case of imposition of extraordinary event, the continuous shift operation staff, depending on the level of severity, would either continued performing activities according the respective intervention instructions and guidelines, or would be gathered in the shelter (Operation Support Centre), in case of imposition of safety measures, from where they would perform the required interventions on the technological equipment or provided operational support to HZSp with respect to recovery and rescue works.

In case that flow paths from CNTN to the reactor building are opened, the gas discharged from I.O and containing a significant volume of fission products could leak into the reactor building. Likewise, the coolant pumped using emergency make-up systems with an isolated CNTN, which use the CNTN sump as a source, contains a high share of fission products.

During containment spraying, the fission products could penetrate the reactor building through normal leaks of the spraying system. If, at this time, it is necessary to perform local handling operations, the necessary protective measures would be taken for the members of local emergency response groups, because the habitability of certain parts of the reactor building could be limited.

Each shelter in the NPP is provided with equipment that facilitates the protection of persons against the effects of radioactive substances, combat poisons and combat biological products. Structurally, these shelters are designed to provide protection to persons against the effects of initial radiation. The technical equipment of the shelters allows their operation for at least 72 hours. The basic furnishings of the shelters include radiation monitoring devices for measuring surface contamination and dose rate, an inventory of spare protective equipment, spare clothes, iodine prophylaxis, communication means. The distribution of spare emergency protective equipment, spare clothes and medical materials is provided by the members of the shelter group based on reasonable needs and requirements of the sheltered persons.

Directly in the NPP, no heavy equipment to clear the backbone and access communications, which might be blocked by debris of civil structures not having sufficient seismic resistance, is available. This might make the access of mobile equipment to the main generating units more difficult. Utilization of the means through IZS has been established.

6.1.3.4 Impact on the accessibility and habitability of the main and secondary control rooms, measures to be taken to avoid or manage this situation

MCR and ECR are located in the clean section of the reactor building. This section could be affected by radiation during large releases of fission products from the containment. However, MCR and ECR are equipped with filtered air-conditioning systems supplied from emergency power supply systems and hence the habitability of these areas is ensured also in case of supposed fission products releases.

If it is impossible to use MCR, the operating management personnel would move to ECR, from which it is possible to monitor the operating parameters and control the components of technological systems to the same extent as from MCR.

If the conditions in MCR get worse, the operators would conduct activities aimed at extending the possible time of stay of the personnel in MCR, depending on the specific situation (e.g. shut-down of supply air-conditioning means, start-up of filtered air-conditioning systems using aerosol and iodine filters) and, if needed, would use radiation protection means (protective clothes, breathing apparatuses, etc.).

After damage to reactor core, fission products could penetrate the reactor building and the accessibility and habitability of certain rooms in the reactor building could be limited. If, at this time, it is necessary to perform local handling operations in these rooms, the necessary protective measures are prepared for the members of local emergency response groups. This particularly includes radiation protection means, shielding protection, distance protection, limited time of stay, etc.).

6.1.3.5 Impact on the different premises used by the crisis teams or for which access would be necessary for management of the accident

In case of an extraordinary event, all of the necessary activities would be managed and carried out from protected locations. The Technical Support Centre, which manages strategies in accordance with SAMG together with Sfift Suprvisor and Emergency Commision, is situated in the Emergency Control Centre. Remote activities for implementing strategies would be carried out by operating management personnel from MCR or ECR. Local activities and repairs to the equipment in respective clean sections of the reactor building, turbine hall or outside structures, if any, would be provided by emergency response groups stationed in the Operating Support Centre (after risk specification, under precisely defined conditions and limitations).

The Emergency Control Centre is a secured workplace with the possibility of permanent habitability for at least 72 hours without off-site support. Emergency Control Centre is equipped with its own power supply sources, filtered ventilation and the possibility of isolation from the external environment. A reserve of water and frozen food is prepared there. The environment in emergency control center is monitored by means of dosimetric equipment and, if the specified values are exceeded, an isolation regime is activated.

6.1.3.6 Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods)

For safety function fulfillment no threats caused by external hazards are identified.

- To ensure the performance of safety functions, procedures and strategies are prepared for the stage preventing damage to reactor core (EOPs) as well as for the stage after damage to fuel in reactor core (SAMG). Thanks to a symptomatic approach to the settlement of emergency conditions, their applicability is not limited due to external conditions. Procedures to determine the degree of damage to the equipment after a seismic event are prepared.

- EOPs and SAMG do not consider any mobile or non-technology means and supplies. The use of supporting and alternative technical means, if any, would be settled on case-by-case basis using OHO mechanisms. Documentation for accident management in EC and TSC is based on the condition of accessibility of data in Emergency Control Centre and in TSC. However, it is not prepared for the purposes of activation of EC and TSC in other locations.
- Plant personnel is sufficiently qualified and trained to use EOPs and SAMG as well as to perform an evaluation of damage to the equipment after a seismic event. The personnel is equally trained to perform handling operations to ensure supply from on-site and off-site sources in case of SBO.
- Deficiencies involving the number of personnel needed to mitigate the above-mentioned consequences of beyond design basis events are not identified within the framework of the shift (IOHO) and POHO.

In case of an extensive destruction to the infrastructure and long-term inaccessibility of the site (collapsed buildings, damage to roads, etc.) problems with support in extraordinary events consequences mitigation could appear:

- The rotating personnel might not reach the site during the initial several days. In that case, the required activities would have to be secured by the personnel present on site during the event. Rotation would be settled in cooperation with public authorities (IZS, army, etc.).
- It would not be probably possible to use emergency preparedness shelters or workplaces of the Emergency Commission or the Technical Support Centre, which are located under structures with a low seismic resistance and are not protected against flooding. The inaccessibility of the Technical Support Centre could be partly solved by its operation from the control room or emergency control room (so far there are no detailed instructions available). Inaccessibility of shelters, if any, would be settled case-by-case by evacuating the non-essential personnel outside the site.

6.1.3.7 Unavailability of power supply

An assessment of power supply unavailability is given in chapter 5.1.

The limited capacity of batteries for category emergency power supply might not allow all interventions at an early stage of a severe accident and could render some measurements inactive. This period could be extended by the controlled disconnection of non-essential consumers. The scope and sequence of equipment and components to be disconnected in order to reduce battery load depend on their importance in relation to the ongoing accident event and the strategy deployed. The goal of disburdening the batteries would be to ensure the longest possible time of operation of I&C systems (control and parameter monitoring) and the supply of equipment needed to perform the necessary safety activities (start of DG and renewal of supply, isolation of coolant discharge routes from RCS, pressure control in SG and RCS, isolation of containment, etc.).

Emergency lighting would be operational until the batteries are fully discharged. Loss of lighting could contribute to the personnel's worse orientation, i.e. to a longer duration of handling operations. In case of loss of supply to both units, the shift personnel could be overburdened with activities to renew supply.

6.1.3.8 Potential failure of instrumentation

All of the required information on the condition of components and values of parameters needed to cope with severe accidents is available in PAMS and are either directly processed in PAMS or transferred from other I&C systems of safety systems.

All systems providing information on safety-related variables in PAMS and PAMS as such are qualified for accident conditions and post-accident conditions and are supplied from the batteries of emergency power supply systems. Even though these systems are qualified for design-basis accident and post-accident conditions, the requirements for managing severe accidents were also taken into account in the design of these systems to a reasonable extent. For example the range of temperature measurement at the core exit is up to 1300 ℃ or the range of measurement of hydrogen concentration in the containment is up to 10 %.

A limited set of parameters is used to diagnose the accident condition and verify the implementation of selected strategies. The measured values of selected variables from standard instrumentation serve to verity these parameters. For each parameter, several variables are determined, through the use of which the specific parameter (size, trend) can be verified. A direct measurement of the required parameter and one or several measurements of alternative variables are always used; they can be used to infer the size or trend of the required parameter. In several cases it is not possible to evaluate the size or trend of the required parameter in case of a severe accident on the basis of the directly measured values either for reasons of their unavailability or absence of measurement of the specific parameter. In such cases, computational aids are used to determine the required parameter (simple charts showing the dependencies of these parameters). The inputs for these computational aids can come either from directly measured values or from pre-determined, defined values.

The sensors and converters within the containment will be, in accordance with the design, operable in the long term at a temperature of up to 150 ℃ and an overpressure of up to 0.4 MPa. Follow-up systems for signal processing located in I&C cabinets will be operable, in accordance with the design, until the temperature of 40 ℃ is reached in the specific rooms.

6.1.3.9 Potential effects from the other neighboring installations at site, including considerations of restricted availability of trained staff to deal with multi-unit, extended accidents

There are three branches of a transit gas pipeline passing at a minimum distance of 900 m from NPP units. It was established for a simultaneous rupture of all three gas pipelines across their full cross-section, with the ensuing gas outburst that would end up in flames, that the effects would not affect the equipment that provides for safety of NPP units.

In case of gas leak from the gas pipeline, there are sensors to detect a concentration of natural gas, if applicable, installed on selected openings of the air-conditioning systems. If natural gas could spread underground, e.g. in case of frozen soil (or under snow or ice cover), the concentration of natural gas is detected using measurement sensors located in specially designed measurement points at the boundary of the power plant facing the transit gas pipeline. Signaling is derived from these sensors and the applicable activities are determined if a concentration of natural gas is detected. An adequate response in this case is also incorporated in the emergency preparedness system.

Other risks due to on-site causes, resulting from the operation of the chemicals storage, the technical gases storage and fuel oil management, were also assessed. These risks were identified as negligible.

6.1.4 Conclusion on the adequacy of organizational issues for accident management

Temelin NPP has implemented a severe accident management system to ensure level 4 defense in depth and an emergency preparedness system to ensure level 5 defense in depth. The functioning and interconnected accident management and emergency preparedness system at Temelin NPP is based on a robust set of personnel, administrative and technical measures. For the personnel area, it is the existence of an emergency response organization and activities related to the individual functions. For the administrative area, it is the implementation of relevant procedures, guidelines and instructions using the capacities of support centers, and for the technical area, it is the functionality of the required scope of technical means for implementation of strategies. Interventions (actions) during the first (preventive) stage of an abnormal event are taken by the personnel of continuous shift operations. Stage two (mitigation of consequences) begins if the scope of event is getting beyond the capacity of the shift personnel where the emergency response organization is activated. In this case, the responsibility for management of interventions (actions) is taken over by the emergency staff of Temelin NPP with the assistance of the technical support centre.

In case of an abnormal event, all the necessary activities would be managed and performed from places protected against the effects of activity leak into the environment. The technical support centre and emergency staff who are responsible for management of strategies according to SAMG are based in the emergency control center. It is a protected workstation enabling permanent occupancy for a period of at least 72 hours without external support. Remote activities to implement strategies would be performed by the shift personnel from MCR or ECR. Local activities and possible equipment repairs in the relevant cold area of the reactor building, turbine hall or support buildings would be provided by intervention teams based in the operating support center.

The concept of the accident management system at Temelin NPP is based on symptomatic approach. Strategies are developed for the management of technological accidents (up to fuel failure) and included in EOPs, with the highest priority to recover heat removal from the core and to prevent damage on the first barrier against leak of fission products (fuel cladding). Strategies are developed for mitigation of consequences of fuel failure accidents (severe accidents) and included in SAMG, with the highest priority to prevent damage on the third barrier against leak of fission products (containment) which at that time is the last intact barrier. EOPs and SAMG are updated on a regular basis to incorporate knowledge from training of their use on the simulator, or emergency exercises as well as external knowledge. Currently is being prepared development of Severe Accidents Management Guidelines for shutdown states (SSAMG), which are intended to manage events when a beyond design basis situation due to fuel damage in reactor core develops into a severe accident under shutdown unit (with open reactor), or to manage beyond design basis events due to fuel damage in SFP.

The emergency preparedness system is implemented in case of safety challenge to the reactor (unit) or site area, or under circumstances which the shift personnel are unable to manage. The emergency response organization (OHO) would be activated in case of any level of abnormal event (Alert, Site emergency, General emergency). Its internal part (IOHO) consists of shift personnel and its emergency part (POHO) consists of NPP technical staff experts who are on call duty.

A system of qualification requirements has been implemented to select shift personnel and experts for POHO and additional criteria are considered to take their knowledge and expertise into account. The readiness of shift and technical personnel to manage technological accidents is tested on a periodical basis during full-scope simulator training with the participation of the Technical Support Center staff and during emergency exercises. It is necessary to adopt additional measures in particular to improve selection of personnel with the best knowledge in the area of severe accident management.

The organization of abnormal event (emergency) management (including severe accidents) is defined in the on-site emergency plan approved by SONS (State Office for Nuclear Safety).

If emergency conditions occur (design basis or beyond design basis event without fuel failure), all currently available technical means within their scope of design specification will be employed to meet the EOP requirements. SAMG envisage to perform the required actions using all available systems and components, or all available technical means even outside design specifications.

There is a corporate fire brigade unit available on the Temelin site which disposes of the relevant fire protection equipment and they are trained to intervene at any point of the site area. The pumping equipment of the fire brigade belongs to the major mobile non-technological means available for transport and pumping of fluids.

The accident management program at Temelin NPP is analytically supported for a long term. Analytical support is based on probabilistic and deterministic approach, consisting in selection of the most likely emergency scenarios leading to severe accidents, and consequently in their deterministic analysis using integral computation codes. The result of analytical support is a summary of knowledge based on understanding of phenomena during severe accidents and their timing, identification of possible project weaknesses, and determination of activities to mitigate consequences of severe accidents, and validation of severe accident response activities, and determination of the source term to evaluate possible radiological consequences. Analytical support will also produce a simulation tool to display phenomena during scenarios of severe accidents.

With regard to the specified facts the area of accident management and emergency preparedness is evaluated as complex and sufficient without any significant gaps.

6.1.5 Measures which can be envisaged to enhance accident management capabilities

The provisions for accident management to provide the level 4 of the defense-in-depth and emergency preparedness to provide the level 5 of the defense-in-depth are established fir the cases that all design provisions for levels 1, 2 and 3 of the defense-in-depth fail in parallel.

Although there are several diverse systems available for implementation of each strategy of accident management, areas for additional improvement of safety were identified in the area of severe accident management. Availability of proposed additional technical means and implementation of instructions for their use in fulfilling the safety functions (in case of loss of design systems) would improve capability of the plant to manage development of beyond design basis scenarios to the area of severe accidents.

- In the area of technical preparedness, the sufficiency of alternative technical equipment to provide for the performance of safety functions in case of loss of all design bases SSC is concerned. Nevertheless, based on the previously identified knowledge modifications of the design are being prepared to improve the resistance of the units to the SA consequences. This is related to the fact that with current design capabilities the possibility of containment integrity threat by hydrogen produced after severe accident can not be totally excluded. In addition there are only limited possibilities of how to provide containment integrity after reactor vessel bottom damage on long term basis.
- In the area of administrative management, this concerns in particular the Severe Accidents Management Guidelines for shutdown states (SSAMG), which have not been finalized yet. Nevertheless, for the purpose of EOPs, SEOPs and SAMG, the so-called “Maintenance program”, including the update thereof, is performed on a regular and systematic basis.
- In the personnel area, problems concerning accessibility of the plant or the Emergency Control Centre utilization may appear, which entails problems with management, decision-taking with respect to high-risk variants of the solution within the accident situation management and but not least, with communication and warning for the staff members.

To improve the accident management system, measures concerning the following areas are to be elaborated on:

1. Organizational provision to reach more effective utilization of the existing capacities or definition of additional capacities - the so-called crisis plans to manage unpredictable

situations occurring in NPP (the whole location affected, loss of emergency preparedness control centers, loss of communication and warning systems, decision-taking concerning high-risk variants of the solution, altering members of the staff, extreme natural conditions, ...).

2. Completion of some technological instructions / procedures / guidelines to manage the selected beyond design basis states and severe accidents (SAMG for shutdown states, SAMG for damaged fuel in SFP, EDMG, …) with a view to securing cooling and heat removal from core and SFP and eliminating releases.
3. Additional technical measures to provide for non-technical support functionalities (accessibility of the civil structures, availability of the fire-suppression equipment, provision of Emergency Control Centre and shelters...).
4. Alternative means to secure long-term functional communication between all components of the accident management system (the both on-site and off-site).

See the following table for areas for defense-in-depth improvement in case of events which may result in loss of capability to fulfill safety functions. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation.

Some measures (identified as "PSR finding" in the note) would be implemented regardless of the targeted evaluation, which confirmed efficiency and correctness of previously adopted decision to implement the particular measures for improved resistance of the original design. For detail description of the corrective measures see the Attachment.

Areas for improvement	**Corrective Action**	**Term** (short-term I / middle-term II)	**Note**
Technical means	Alternative containment sump water make up	II	PSR finding
Technical means	Hydrogen disposal system inside containment for severe accidents	II	PSR finding ZKZ B462
Analysis	Melt localization out of reactor pressure vessel	II	PSR finding With other VVER

Areas for improvement	Corrective Action	**Term** (short-term I / middle-term II)	Note
Technical means	SSC functionality verification in out of the design operating conditions	II	
Analysis	Analysis of radiation situation in MCR/ECR during severe accidents		
Procedures / guidelines	Development of „shutdown SAMG“ (fuel damage with open reactor / in SFP)	II	PSR finding Start in 2012
Procedures / guidelines	EDMG instructions for use of alternative means	II	
Personnel	Emergency organization attendance by qualified and trained personnel	I	
Emergency preparedness	Capability of emergency organization to operate out of Emergency Control Center	I	
Emergency preparedness	Prepare agreements with external forces (IZS, army) in vicinity of NPP. Organizational measures.	II	

6.2 Accident management measures in place at the various stages of a scenario of loss of the core cooling function

6.2.1 Before occurrence of fuel damage in the reactor pressure vessel (including last resorts to prevent fuel damage)

The measures to manage accidents in case of loss of cooling (before damage to fuel) are described in EOPs, and there are sufficient diversified technical means to implement the measures.

A degradation of core cooling is identified when the temperature of 370 ℃ is exceeded at the core exit. With this temperature, the critical temperature of water at the core exit is exceeded and the steam enters the stage of overheated steam. This is therefore a clear symptom of core uncovery. If core cooling is not restored, the temperature of 650 ℃ at the core exit will be exceeded due to insufficient core cooling. An analysis of the conditions of insufficient core cooling implies that when this temperature is reached, there has already been core uncovery to the extent of about three-fourths. Heat removal after core uncovery is provided using the remaining coolant and, in particular, through the generated steam, which gradually overheats by passing the exposed fuel clusters, depending on the degree of core uncovery. A three-fourth core uncovery shows in a decrease in steam production in core, which also results in a decrease in the volume of coolant that returns to core after condensing in SG. This leads to an increased trend in temperature rise at the core exit. In case of a further rise in the temperature of overheated steam in core, the conditions would be established for an intensive development of steam-zirconium reaction of the cladding, as another major source of heat. The first barrier against the leak of fission products would be exposed.

The strategies for managing accidents related to a loss of core cooling are contained in EOPs at the stage preceding severe damage to fuel in the core and in SAMG at the stage after severe damage to fuel in the core. A consistent symptomatic approach in SAMG is advantageous in terms of the primary goal - protection of the containment against damage.

Core cooling at the stage preceding severe damage to fuel is recovered using the activities described in EOPs. There are several strategies defined for recovering core cooling at this stage.

- The recovery of high-pressure RCS charging (high-pressure emergency core cooling, emergency boration, normal RCS make-up, etc.) for recovery core cooling.
- SG depressurization - increase of heat transfer from RCS to SG, which results in steam condensing on the primary side of SG pipes. When the speed of condensing exceeds the speed of steam generation, the RCS pressure will start decreasing, which may ensure coolant charging to RCS from low-pressure sources. This RCS depressurization may ensure the accumulator injection into RCS.
- RCS depressurization - reduction of RCS pressure may ensure the coolant charging to RCS from low-pressure sources. Even though this is an effective method, its disadvantage is another loss of the already limited volume of coolant in RCS in case the purpose of depressurization - actuation of low-pressure systems - is not met. At any rate, depressurization means that the injection of cold coolant from accumulators is allowed.
- The use of remaining coolant inventory in RCS - equilibrate of levels between core and the reactor downcomer by connecting the upper parts of the RCS equipment and

the cold legs of circulating loops, an attempt to actuate RCP for the possible cooling of core by transporting water from the loop seal created in a section of the cold leg at RCP suction. The RCP start-up will also cause circulation of overheated steam.

6.2.2 After occurrence of fuel damage in the reactor pressure vessel

The measures to manage accidents after severe damage to fuel are described in SAMG strategies, which use all available RCS charging means to recovery core cooling. Each particular RCS make-up system is able to supply a sufficient coolant inventory for decay heat removal from the damaged fuel, even though flooding of RPV from inside does not guarantee core cooling because this can, due to melting, enter a state where cooling is no longer possible.

All strategies are based on the principle of damaged fuel cooling from inside RPV, i.e. by making up water to RCS. Taking into account the reactor's thermal power and the design-basis solution of the concrete reactor cavity, there is no possibility for VVER 1000 units with V320 reactors to ensure the cooling of RPV from outside.

If core is not damaged and is flooded with water, core cooling is sufficient to prevent its damage. If there is no water in core, the decay heat is absorbed in the materials of the core. If water make-up to RCS is not initiated, the heating of core continues.

Heat removal after severe core damage is recovered by actions described in SAMG. For heat removal recovery the following strategies are identified:

- Make-up water to the hot, dried core always positively influences the progress of the accident. An optimal method of RCS make-up recovery is established so as to minimize the ensuing leakage of fission products to the atmosphere. If the flow rate of the make-up water is sufficient to ensure energy removal at a higher rate than the production of decay heat, it is possible to recover core cooling.
- Another measure after serious damage to fuel is RCS depressurization. The aim of RCS depressurization is to reduce RCS pressure below a value for which it has been demonstrated that direct containment heating is no longer possible because the molten core will not be ejected from the reactor under high pressure. There are several methods for RCS depressurizing (use of an emergency RCS venting system, a PRZR relief valve, normal PRZR spray, SG depressurization etc.).

6.2.3 After failure of the reactor pressure vessel

When RPV fails, the core debris will move to the concrete reactor cavity or other parts of the containment. If there's no water in the containment, the core debris will start attacking the concrete bottom of CNTN and a Molten Core Concrete Interaction (MCCI) will follow. The consequences of this reaction include production of hydrogen and other non-condensable gases. The measures to manage accidents after severe damage to fuel and relocation of the melt to the bottom of the containment are described in SAMG strategies, which use all available make-up means to the containment to cool the melt

All strategies for cooling the melt at the bottom of the containment are based on the principle of spaying the melt from above. The flooding of core debris outside RPV with water will ensure heat removal from the debris and reduce the velocity of the concrete attack.

One of the outputs from analyses of sequences resulting in severe accidents, selected on the basis of PSA2 results, is also the time until RPV integrity is lost due to effects from the core melt. This time can be used in the most unfavorable scenario provided that all methods of coolant supply to the RPV failed, approx. 4.5 hours. This moment characterizes the conclusion of the in-vessel phase of a severe accident and the beginning of the ex-vessel phase, associated with all negative phenomena in the containment (molten core concrete interactions with hydrogen production, direct containment heating etc.).

The RPV failure after serious damage to fuel, which ends the in-vessel phase of a severe accident, has the nature of cliff-edge effect conditions from the point of view of negative phenomena in the containment.

Make-up water to the containment is a preventive measure in case of a severe accident. The relevant strategy in SAMG provides instructions for CNTN flooding up to the maximum measurable level, for which protection of the concrete at the bottom of CNTN is ensured in case of leakage of core debris from RPV to the containment and an efficient washing-up of fission products leaking from the melt is provided.

If the molten core-concrete mixture is flooded with water, the heat transfer from the upper surface of this mixture will be significantly more efficient due to water boiling. Water will also penetrate cracks and get between fragments created on the surface of the crust, which further improves heat removal from the molten core pool.

Make-up of water to CNTN during a severe accident is associated with several expected positive effects:

- Water at the bottom of CNTN will remove heat from the core debris and will reduce molten core concrete interactions.
- Water from the CNTN sump may be used for operation of ECCS pumps and CNTN spray pumps.

- The fission products releases from the core debris, located at the bottom of CNTN, will be washed up by the water layer.

During a severe accident, the containment can be fed with water using standard methods of make-up from re-fuelling water storage tanks or from primary coolant drainage tanks and also an alternative method of make-up using stationary fire fighting pumps or by overfilling the PRZR relief tank.

6.3 Maintaining the containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core

6.3.1 Elimination of fuel damage / meltdown in high pressure

6.3.1.1 Design provisions

The basic design-basis tools to ensure RCS depressurization include an emergency RCS venting system, PRZR relief valve, normal PRZR spraying and SG depressurization. However, there are limiting conditions for their use in case of beyond design-basis accidents.

RCS depressurization as a preventive measure to prevent the possibility of fuel damage under a higher pressure is already used in EOPs as a response to inadequate core cooling. If all these activities aimed at reducing the RCS pressure fail, SAMG contains instructions to evaluate the availability of equipment necessary for RCS depressurization and for mitigating the negative impacts associated with this depressurization. The aim of RCS depressurization is to reduce RCS pressure below a value for which it has been demonstrated that direct containment heating is no longer possible because the melt will not be ejected from the reactor under high pressure.

6.3.1.2 Operational provisions

Reducing RCS pressure is one of the key priorities for managing severe accidents. There are several methods for depressurizing RCS:

- Use of an emergency RCS venting system.
- PRZR relief valve.
- Normal PRZR spraying.
- SG depressurization.

Reducing RCS pressure during a severe accident is associated with several positive impacts:

- Reduction of the possibility of high-pressure melt ejection (HPME).
- Reduction of the possibility of creep damage to SG tubes and SG primary manifold (reduced possibility of releases outside the containment).
- Allowing RCS make-up from multiple sources (especially low-pressure).

6.3.2 Management of hydrogen risks inside the containment

6.3.2.1 Design provisions, including consideration of adequacy in view of hydrogen production rate and amount

Two hydrogen combustion regimes are the most dangerous for containment integrity - rapid deflagration and rapid deflagration to detonation transition. To evaluate the hydrogen risk, the progress of hydrogen spreading and distribution during severe accidents was analyzed for the whole containment area.

The containments are equipped with a system for disposal of post-accident hydrogen, designed only for design-basis accidents. This system contains passive auto-catalytic recombiners and is able to dispose of hydrogen released during accidents and in post-accident conditions only for design-basis accidents in the long term, thus maintaining the hydrogen concentration at values where its combustion is not possible. The existing hydrogen disposal system may not be sufficient for severe accidents. At present, however, design preparation is underway for the installation of a hydrogen disposal system to dispose of hydrogen generated during severe accidents.

Containment venting (filtered or unfiltered) is potential possibility to decrease hydrogen amount in the containment using only systems that that are not primarily intended for venting in the design.

6.3.2.2 Operational provisions

The current measures for managing accidents in case containment integrity is exposed by hydrogen are described in SAMG strategies, which use all available means to prevent dangerous forms of hydrogen combustion. In addition to the strategy of intentional burn of hydrogen, which has a limited field of application and is based on random hydrogen inflammation using an electric consumer in the containment, all other strategies for preventing dangerous forms of hydrogen combustion consisting in establishing conditions where hydrogen flammability is mitigated. The strategies do not result in actual hydrogen disposal, but only in limiting the conditions for its flammability. The use of all existing options therefore does not fully eliminate the possibility of exposing the containment due to hydrogen, but extends the timeframe for the operation of the current system of recombiners.

As the last option for preventing damage to containment integrity, the strategies described in SAMG stipulate controlled venting of the containment using systems that are not primarily intended for venting in the design. SAMG strategies, however, deal also with the possible negative impacts, in this case with the leakage of hydrogen outside the containment.

6.3.3 Prevention of overpressure of the containment

6.3.3.1 Design provisions, including means to restrict radioactive releases if prevention of overpressure requires steam / gas relief from containment

The containment's design function is to limit the potential radiation consequences of an accident to RCS. This function is, among others, provided by the containment structure, which limits leakage outside the containment to very low values even at a high internal overpressure inside the containment. Since the whole RCS pressure boundary is located inside the containment, the containment acts as the last barrier against the leakage of radionuclides that can be released from the fuel or RCS coolant in case of an accident.

The containment integrity is ensured in the design using the following systems:

- Containment isolation system – isolation valves automatically closed in case of pressure increase in the containment - operability conditional on power supply availability.
- Containment pressure reduction system - spray pumps and storage tanks with chemical agents to trap post-accident iodine - operability conditional on power supply availability.
- Post-accident hydrogen disposal system - passive auto-catalytic recombiners, designed for design-basis accidents - do not require power supply.

The design systems for reducing pressure include 3 trains of a spray system (3 x 100%), each capable of reducing pressure in the containment by means of condensing the steam generated when the piping ruptures (up to the primary circulation loop piping rupture. The design-basis load of the containment in case of the so-called large LOCA is less than 150 ^{o}C and 0.49MPa.

Containment pressurization is one of the dominant ways of breaching containment integrity, which potentially leads to the release of larger quantities of fission products to the atmosphere. Containment pressurization during a severe accident could occur due to dynamic phenomena (i.e. hydrogen combustion) or long-term accumulation of steam or non-condensed gases in the containment atmosphere. Dynamic phenomena may cause pressure peaks, which may not be mitigated through normal heat removal from the containment (i.e. the energy increase in the containment is higher than the removal capacity of the spray systems).

Analyses were performed to determine the limit case of pressure increase in the containment. The analyses imply that by the time of RPV failure and penetration of the debris to the bottom of the concrete reactor cavity, the pressure in the containment cannot increase to the values that would mean a serious exposure of its integrity. Only when the MCCI starts in the ex-vessel phase, there could be additional pressure increase in the containment above the values that expose its integrity.

The containment strength calculations show that when the design-basis pressure value in the containment is exceeded, the containment structure displays linear behavior first. Only later there will be cracks appearing on the inner side of the concrete, gradually leading to the plasticization of the steel lining and ultimately to the breach of containment tightness.

Reaching the pressure value for containment integrity breach (approximately 1.6 times the design-basis pressure value) has the nature of limit cliff-edge conditions in terms of exposure to containment integrity due to pressurization.

6.3.3.2 Operational and organizational provisions

EOPs envision the use of CNTN spraying so that the pressure inside CNTN remains within the design-basis parameters. After an accident occurs, the CNTN spray system is able to ensure at least decay heat removal. Therefore, operation of the spray system should, from a long-term point of view, result in maintaining the pressure in CNTN at a value equal to the ambient atmospheric pressure, unless there was a major release of non-condensed gases while the molten core reacted with the concrete.

At the stage preceding serious damage to fuel in core as well as the stage following serious damage to fuel in core, the proposed strategies are also aimed at preventing a containment by-pass or minimizing radioactivity release in case of breach of the heat-exchange surface in SG or leak through ECCS heat exchangers to ECW.

The measures for managing accidents in case containment integrity is exposed by high pressure are described in SAMG strategies, which use all available means to reduce the pressure in the containment. The applicable strategies in SAMG provide instructions for carrying out preventive measures to reduce pressure in the containment when its integrity is exposed to pressurization:

- Containment spray systems (normal containment spraying or fire water)

  Containment spraying using fire fighting pumps is an alternative method (if power supply is available). Taking into account the layout of fire fighting pumps and the corresponding tanks, sufficient diversification is provided compared to containment spray systems.

- Containment air-conditioning units (with coolers)
- Different (non-design-basis) paths for filtered and, if applicable unfiltered venting

Containment venting using systems that are not aimed for venting in the design is identified as one of the many activities to mitigate any serious exposure of the containment due to high pressure. Containment venting reduces the pressure in the containment at the cost of release of fission products to the atmosphere. Due to the release of fission products, containment venting is considered as the ultimate option to prevent containment failure.

6.3.4 Prevention of re-criticality

6.3.4.1 Design provisions

In case of RCS make-up by water with a low content of H_3BO_3 and reduction of the RCS boron concentration, re-criticality may be reached along with an increased reactor power due to increased neutron moderation, if the original core geometry is maintained. If the original core geometry, i.e. the ability of neutron moderation, has been already lost, criticality cannot occur. If the reactor returns to power, it does not mean any immediate risk because it is limited due to the formation of bubbles in core.

After serious fuel damage and loss of control rods geometry, a cavity could be created without any absorbers. Due to the absence of a moderator (water always changes into vapor in these conditions), the ability of moderation will be lost. Because of the breached core geometry, criticality cannot occur to a larger extent.

6.3.4.2 Operational provisions

Measures to prevent the reduction of boron concentration have a top priority during activities in accordance with EOPs at the preventive stage before fuel damage, when the original core geometry is fully maintained, allowing for neutron moderation and occurrence of a criticality. When activities are performed in accordance with SAMG, at the stage following serious fuel damage and the loss of original core geometry, the respective strategies describe measures if the rate of power increase rises, but considering the impaired core geometry, a criticality cannot occur to a larger core extent.

6.3.5 Prevention of basemat melt through

6.3.5.1 Potential design arrangements for retention of the corium in the pressure vessel

Taking into account the reactor's thermal power and the design-basis solution of the concrete reactor cavity, there is no possibility for VVER 1000 units with V320 reactors to ensure the RPV cooling from outside.

6.3.5.2 Potential arrangements to cool the corium inside the containment after reactor pressure vessel rupture

When RPV fails, the core debris would move to the concrete reactor cavity or other parts of CNTN.

The measures for managing accidents in case of containment integrity exposure due to molten core-concrete interactions are described in SAMG strategies. Make-up water the containment is a preventive measure in case of a severe accident. The applicable strategy in SAMG provides instructions for CNTN flooding up to the maximum measurable level, for which protection of the concrete at the bottom of CNTN is ensured in case of leakage of core debris from RPV to the containment and an efficient washing-up of fission products leaking from the melt is provided.

The conducted analyses showed, however, that cooling the pool of molten materials in the cavity with water may reduce the rate of concrete decomposition and hence postponing containment failure to a later stage of the accident.

The flooding of core debris outside RPV with water would ensure heat removal from the debris and reduce the speed of concrete attack. During a severe accident, the containment make-up can be ensured using the following:

Normal operating provisions:

- Make-up from storage water tanks intended for re-fuelling.
- Make-up from primary coolant letdown tanks.

Alternative method of make-up:

- Make-up using fire fighting pumps.
- Overfilling of the bubbler tank.
- If RPV was damaged, RCS make-up will also result in an outflow of water to the concrete

reactor cavity through the hole in RPV, which will ensure cooling of the core debris located in the reactor cavity.

There are also other strategies to manage the melt leaking from RPV in order to slow down the degradation of the containment's basemat, i.e. to postpone or completely stop the massive leak of radioactive substances to the external environment after containment bottom melt-through. These strategies exclude a large-scale steam explosion and, by reducing the thickness of the melt layer, increase the likelihood of melt cooling and suspension of concrete decomposition due to the molten materials.

To improve resistance of the containment bottom in the concrete reactor shaft against melting through by melt after reactor pressure vessel failure a modification has been implemented in form of stuffing of ex-core neutron flux measurement channels in the containment bottom. The channels were filled with removable steel sheaths filled with fireproof material. This modification provides high resistance against melt penetration with no effect on neutron flux measurement instrumentation.

6.3.5.3 Cliff edge effects related to time delay between reactor shutdown and core meltdown

Analyses of the SBO scenario, when a loss of secondary heat removal occurs, imply that without performing alternative activities, described in EOPs, there is a very short time reserve to recover heat removal from I.O. The temperature of 650 ℃ at core exit could be, in the most unfavorable case, reached in approximately 2.5 to 3.5 hours from SBO occurrence. Reaching a temperature higher than 650 ℃ at core e xit, which is constantly increasing has the nature of cliff-edge effect in terms of serious fuel damage in core.

In case of loss of core cooling for long term, RPV integrity could be threaten by molten core. This time can be used in the most unfavorable scenario provided that all methods of coolant supply to the RPV failed, approx. 4.5 hours. This moment characterizes the conclusion of the in-vessel phase of a severe accident and the beginning of the ex-vessel phase

When the core debris moves from RPV to the bottom of the containment the molten core-concrete interaction starts. This interaction results in concrete decomposition, with intensive hydrogen production. The containment bottom is then attacked and when its thickness reaches a value where the residual concrete would be broken by the weight of the melt, the melt will penetrate the lower non-hermetic section of the reactor building. In the most unfavorable case, the melt may penetrate to the lower non-hermetic section of the reactor building about 24 hours after the occurrence of the accident.

6.3.6 Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

6.3.6.1 Design provisions

The integrity of containments is ensured in the design using the following systems:

- Containment isolation system – isolation valves automatically closed in case of pressure increase in the containment - operability conditional on power supply availability.
- Containment pressure reduction system - spray pumps and storage tanks with chemical agents to trap post-accident iodine - operability conditional on power supply availability.
- Post-accident hydrogen disposal system - passive auto-catalytic recombiners, designed for design-basis accidents - do not require power supply.

The provision of the necessary energy to ensure the functionality of these systems is given in 1.3.5, 1.3.6.

6.3.6.2 Operational provisions

Most of the strategies described in SAMG (make-up water to the containment, heat removal, maintaining pressure in the containment) require power supply availability for successful implementation.

6.3.7 Measuring and control instrumentation needed for protecting containment integrity

A basic measurement for the strategies employed in case of containment integrity exposure due to high pressure is pressure measurement in the containment. Redundant pressure measurements in the containment are carried out in PRPS and communicated through PAMS. The PRPS and PAMS systems are qualified for accident and post-accident conditions and are supplied from the batteries of emergency power supply systems. Even though these systems are qualified for design-basis accident and post-accident conditions, the requirements for managing severe accidents (range of pressure measurement in the containment up to 1.6 MPa) were also taken into account in the design of these systems to a reasonable extent. To verify the success of the strategy of mitigating serious exposure due to high pressure, SAMG identifies parameters whose values are communicated through PAMS.

The containments are equipped with a post-accident containment hydrogen monitoring system (PACHMS). This measurement is part of PAMS and its operability is conditional upon the operability of the PAMS system. There are a total of 16 sensors in the containment, covering all rooms where hydrogen can be accumulated. The PACHMS system measures hydrogen concentrations in a range of 0 ÷ 10%.

No active functions are derived from the post-accident containment hydrogen monitoring system and the post-accident hydrogen disposal system for design-basis accidents also burns hydrogen independently of this measurement. The outputs from the post-accident containment hydrogen monitoring system are used to manage accidents (emergency condition diagnostics, choice of strategy to manage hydrogen flammability, establishment of negative impacts due to hydrogen combustion, etc.).

The PACHMS system and the PAMS system as such are qualified for accident and post-accident conditions and are supplied from the batteries of emergency power supply systems. Even though these systems are qualified for design-basis accident and post-accident conditions, the requirements for managing severe accidents were also taken into account to a reasonable extent in the design of these systems.

Even though post-accident containment hydrogen monitoring provides essential inputs for accident management, it is possible, using the computational aids in SAMG, to estimate the amount of the produced hydrogen based on the development of the accident condition even without any knowledge of the measured values.

Core exit temperature measurement is used for evaluating the loss of core cooling. The core exit temperature is measured using thermo-couples and the range of this measurement is up to 1,300 ℃. The measurement is part of PAMS and is qualified for accident and post-accident conditions. Core exit temperature measurement is sufficiently redundant. There are 95 thermo-couples installed at the core exit, plus 3 thermo-couples under the reactor upper head. The cold ends thermo-couples temperature is measured with redundant resistance thermometers.

There is no direct measurement for evaluating the damage to RPV and the relocation of the melt to the containment. At this accident stage, it is possible to monitor the ongoing phenomena only indirectly, on the basis of the parameter values in the containment environment. The success of the containment flooding strategy can be verified especially on the basis of containment level measurement, which can be monitored either on the basis of level measurement in the dirty part of the containment sump, or on the basis of level measurement in the containment. To verify the success of the containment make-up strategy, SAMG identifies parameters whose values are communicated through PAMS.

There is no direct measurement to identify interactions between the melt and the concrete on the bottom of the containment. At this accident stage, it is possible to monitor the ongoing phenomena only indirectly, on the basis of the parameter values in the containment environment, which are available in PAMS. These measurements include the measurement of containment pressure and level, hydrogen concentration in the containment and RCS pressure. Using the values from these measurements, it is possible to derive indirectly, using an applicable computational aid in SAMG, if molten core-concrete interactions are possible or are not likely under the specific conditions.

Basic measurements in case of a major fission products release are measurements of dose rate and activity. To measure dose rates and activity, it is possible to use dose rate measurements inside and outside the containment and dose rate and activity measurements in ventilation stacks as well as measurements from a teledosimetric system installed on the site boundary. The ranges of all of these measurements are designed for operating as well as for accident and post-accident conditions.

Radiation measurements with the ranges for accident and post-accident conditions are performed in PAMS and all radiation measurements are communicated through PAMS. Even though these systems are qualified for design-basis accident and post-accident conditions, the requirements for managing severe accidents (e.g. the range of dose rate measurement in the containment up to 10^5 Gy/h) were also taken into account in the design of these systems to a reasonable extent.

6.3.8 Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

Both units are mutually independent in terms of technology and separated in terms of construction design. Accident management activities for both NPP units are controlled from Emergency Control Centre (TSC and Emergency Commission) and interventions in the individual units are performed by the staff personnel of the respective unit. Depending on the current situation on the individual units, the capacities can be relocated between the units as needed. If there is an accident on one of the units, the TSC personnel has specific instructions for a decision on the mode of operation and performance of the necessary activities on the neighboring unit. If the event develops into a severe accident on both units, the same SAMG instructions would be used for both units, but the situation in the individual units would be evaluated separately and TSC and Emergency Commission would carry out the necessary coordination between the activities conducted on both units.

6.3.9 Conclusion on the adequacy of severe accident management systems for protection of containment integrity

SAMG contains strategies developed for mitigation of consequences of severe accidents with the main priority to prevent damage of the 3rd barrier against fission product leakage (the containment) which is the last integral barrier at that moment. All available technical means, including those that are not preferentially designed for severe accident management, are always used to manage beyond design-basis and severe accidents. The use of these means is described in the applicable strategies contained in EOPs and SAMG. The strategies are success-oriented, i.e. one of SAMG's secondary goals is to restore the operability of systems and equipment to the maximum extent possible and the implementation of the specific strategy in any of the described manners leads to success. Success in this case refers to the fulfillment of the main goals of SAMG, i.e. recovery of the unit into a controlled, stable condition and restriction of radioactive releases.

There are several diverse systems to implement each SAMG strategy. Analytical validation showed that even if some strategies are not fully supported by adequate technical means, implementation of SAMG strategies to response to exposures caused by phenomena occurring during severe accidents with the use of existing systems and equipment that are not preferentially designed for severe accidents, leads to successful manage of severe accidents.

Long-term post-accident activities relating to severe accident management consist in the continuation of activities to ensure heat removal and elimination of the occurrence of high-energy phenomena (hydrogen combustion or detonation etc.), depending on the condition of the unit. In this case, it can be very problematic to accurately define the condition of the unit, i.e. to define the possible threats. However, when the unit is restored to a controlled stable condition, the basic condition for SAMG termination is then met. Before terminating SAMG and continuing in long-term post-accident activities, SAMG also describes how to identify the condition of the unit as accurately as possible and how to determine the scope of damage and the long-term risks.

Long-term post-accident activities then switch from the stage of searching for a suitable measure to the stage of ensuring the long-term functionality of the identified and successfully applied measures, i.e. for example ensuring that the alternative water supply source will not fail for any reason whatsoever (loss of supply, exhausted water inventory, component failure). This is also associated with a search for alternatives to measures that have been already implemented successfully, i.e. a search for other measures which, when provided, immediately replace those already applied or are held as back-up in case that the currently implemented measures are lost.

Within the framework of the assessment of containment integrity protection measures for certain beyond design-basis, highly improbable scenarios, the possibility of radioactive media leakage to the environment has been identified due to the threat to the containment integrity by hydrogen generated during a severe accident and due to the limited possibilities to prevent the loss of the containment integrity as a result of melt-through of the containment bottom.

6.3.10 Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

In spite of greatly robust measures in SAMG, intended for accident management to prevent loss of containment integrity, areas for improvement have been identified to increase the ability to maintain containment integrity after serious fuel damage, it is possible to propose and implement additional means to ensure CNTN integrity (i.e. prevent fission products releases) in case of a severe accident. These means namely include containment hydrogen disposal system and provisions for melt localization at CNTN bottom).

See the following table for areas for in-depth protection improvement in case of events which may result in a severe accident. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation.

Some measures (identified as "PSR finding" in the note) would be implemented regardless of the targeted evaluation, which confirmed efficiency and correctness of previously adopted decision to implement the particular measures for improved resistance of the original design. For detail description of the corrective measures see the Attachment.

Areas for improvement	**Corrective Action**	**Term** (short-term I / middle-term II)	**Note**
Technical means	Hydrogen disposal system inside containment for severe accidents	II	PSR finding ZKZ B462

Areas for improvement	Corrective Action	**Term** (short-term I / middle-term II)	Note
Analysis	Melt localization out of reactor pressure vessel	II	PSR finding With other VVER

6.4 Accident management measures to restrict the radioactive releases

6.4.1 Radioactive releases after loss of containment integrity

6.4.1.1 Design provisions

Uncontrolled releases of fission products from the power plant after a loss of containment integrity constitute a threat to the health and safety of the population. The strategies in SAMG aim to avert major releases of fission products. If all strategies aimed at averting the loss of containment integrity fail, emergency preparedness measures shall be activated - see 6.1.2.3.

Major leak is defined as a leak that exceeds 100 microGy/h (the criterion for declaring extraordinary radiation event - general emergency under the On-site Emergency Plan). A leak of fission products at this level corresponds to the breach of all barriers, including the containment (or its by-pass).

The leak could come from one or multiple sources (containment, SG, non-hermetic rooms, ECW).

For evaluation of fission products releases the biological effects of fission product leaks would be accessed. The determination of limit conditions in terms of the biological effects of fission product leaks goes beyond the framework of this evaluation.

6.4.1.2 Operational provisions

When a major leak of fission products is identified, all four sources are evaluated. The activities performed to reduce the releases from one source may affect releases from another source. The strategy primarily focuses on identifying the point of release to assure that none of the activities carried out in accordance with these instructions would result in the worsening of the releases. Preventing the releases of fission products to the surroundings, along with preventing a loss of CNTN integrity, as the last barrier against fission product releases, is the main goal of SAMG. However, SAMG also describes strategies to end or at least to reduce the fission products releases after a loss of CNTN integrity with the use of all available means. If CNTN integrity is challenged and large fission products release is ongoing, the corresponding strategy for fission products releases mitigation is immediately implemented since such strategy has the highest priority over the all SAMG strategies.

The measures for managing accidents after a loss of containment integrity are described in SAMG strategies. The strategies are implemented by the shift personnel from MCR or ECR. If both MCR and ECR are not habitable due to a major leak from the containment, there may be a problem with implementing the strategies in accordance with SAMG due to the inability to operate the components of safety systems.

Fission products releases are decreased using CNTN spraying or the CNTN cooling air-conditioning systems, it is possible to reduce the fission products releases by means of their washing-up and depressurization. Filtered venting is the last possibility to reduce the CNTN pressure and mitigate the fission products releases. Before it is used, however, it is necessary to verify that the leaks will not be in any case worsened.

As regards personnel and population protection, fast mobile monitoring group is established to monitor and evaluate the radiation situation in the affected sectors. The RTARC program system is also available for the purposes of preventive measures declared to protect the population.

6.4.2 Accident management after uncovering of the top of fuel in the fuel pool

SFP is located inside the containments. If spent fuel is stored in SFP, it is necessary to keep an sufficient coolant inventory in it and removal of the generated heat shall be provided from the pool. When SFP is operated in the fuel storage mode, compliance with a water level higher than 792cm is required. If the SFP level drops below 550 cm, the heads of the stored fuel assemblies become uncovered.

In case of a total loss of normal SFP cooling (either due to level drop or after interruption of heat removal), the containment spraying system pumps shall be used for SFP make-up, using lines dedicated for SFP emergency make-up. With the use of this system, it is possible to make-up SFP and to ensure the drainage of coolant from SFP, by means of an overspill, to the bottom of the containment, which ensures heat removal from the spent fuel in SFP in an alternative way through the ECCS cooler. This cooling circuit is independent of the SFP cooling system and provides an alternative way of heat removal from the spent fuel stored in SFP. The heat from SFP can be also removed using only evaporation to the containment and the evaporation can be compensated by make-up through the containment spraying system pumps. If it is impossible to remove the generated heat from the spent fuel in SFP through ECW to the ultimate heat sink, this method is limited by the passive thermal capacity of CNTN from the point of view of long-term SFP cooling.

Emergency SFP cooling using the containment spraying system pumps is alternative way of heat removal from SFP in case of a loss of normal cooling capability.

Calculations were used to analyze the loss of SFP cooling with spent fuel stored in it. The calculations resulted in maximum reached temperatures in SFP with cooling, heat-up rates and the time margins until the saturation temperature is reached and until the heads of the stored fuel assemblies are uncovered after a loss of SFP cooling.

The results of calculations for heat-up rates and the time margins until boiling depend on many circumstances, such as the number of fuel assemblies in the individual sections of SFP (generated heat output), the time after unloading the fuel assemblies from core, the SFP level at the time of loss of heat removal, the initial SFP temperature etc. Based on the performed analyses, it can be concluded that depending on the initial conditions, the temperature increase rate in SFP after cooling interruption is from several degrees Celsius per hour to tens of degrees Celsius per hour and the margin until boiling ranges from several hours to tens of hours. At the maximum heat load of SFP, there will be no damage to the stored fuel assemblies earlier than at a late accident stage after a loss of heat removal from SFP.

The time until exposure of the heads of the stored fuel assemblies has the nature of cliff-edge effect in terms of exposure to the cooling of the stored spent fuel in SFP.

As regards the time for performing the activities aimed at recovery the cooling of spent fuel stored in SFP, the situation is better than in case of loss of heat removal from core, but a long-term loss of heat removal exceeding several tens of hours without alternative SFP make-up could lead to damage of the spent fuel stored in SFP.

6.4.2.1 Hydrogen management

The later stages of a loss of cooling of the spent fuel in SFP can be characterized by water boiling, uncovery of the stored fuel and steam-zirconium reaction in the upper section of the fuel. After fuel uncovery, overheating and oxidization in the air would take place, associated with a significantly higher heat escalation (the released heat is higher during oxidization directly by oxygen than steam), nitridation in the upper section with a lack of oxygen and the subsequent oxidization when the non-oxidized Zr in the bottom section is consumed.

These hydrogen sources are localized in the containment. Hydrogen leak from SFP outside the containment is, on condition of isolation of the lines passing through the containment wall, virtually excluded, which excludes an exposure of essential equipment outside the containment due to hydrogen combustion.

6.4.2.2 Providing adequate shielding against radiation

To ensure shielding against radiation from spent fuel assemblies, the SFP level shall not drop below 783 cm. The location of SFP within the containment ensures that even after a decrease of the SFP level below a value needed for shielding against radiation generated by the spent fuel, there will be no undesirable irradiation of people. Under normal power operation and in hot or semi-hot states, the containment is closed and entry inside is controlled on the basis of a special permit (containing also an evaluation of the radiation situation).

Since the containment air locks can be opened in cold state or during shutdown and there can be personnel in the containment carrying out work associated with an outage, the health of these people would be endangered in this case. Therefore, one of the requirements performed immediately after an emergency situation is identified in the applicable procedures is to ensure the evacuation of all personnel who are in the containment at the time of the event and to close the containment air locks.

6.4.2.3 Restricting releases after severe damage of spent fuel in the fuel storage pools

Technical means to mitigate consequences of fuel damage in SFP are available and strategies consist of the continued make-up of water and heat removal to containment and, if applicable, in isolating the leak from SFP as instructed in EOPs. SAMG for shutdown states and accidents with fuel damage in SFP are not developed yet.

No analyses were conducted for damage of the spent fuel stored in SFP. Taking into account the existence of an alternative method of SFP make-up using the containment spraying system pumps, a long-term loss of heat removal from SFP without a simultaneous loss of heat removal from core is not expected.

With the simultaneous loss of heat removal from SFP and core (considering the location of SFP inside CNTN), the restrictions arising from the loss of heat removal from core have the nature of cliff-edge effect because, in terms of the time for performing the activities needed to recover the cooling of spent fuel stored in SFP, the situation is more favorable than in case of a loss of heat removal from core.

6.4.2.4 Instrumentation needed to monitor the spent fuel state and to manage the accident

The SFP level is a key parameter to evaluate the loss of heat removal from the spent fuel stored in SFP. Until the fuel is covered with a layer of water (even when boiling), decay heat will be removed from the fuel. When water evaporates from SFP and the stored fuel assemblies are uncovered, the fuel assemblies will start overheating. The SFP level and several other parameters, such as the condition of ECW systems and the ECW flow rate to the heat exchanger for SFP cooling, are communicated through PAMS. SFP temperature is also measured. SFP temperature measurement is not, however, qualified for emergency conditions and the value of SFP temperature is not communicated through PAMS.

6.4.3 Availability and habitability of the control room

The activities under SAMG are controlled by TSC, which is located in the Emergency Control Centre under administrative building. The activities for implementing the strategy are carried out by staff personnel from MCR or ECR. Local activities and repairs to the equipment, if any, are carried out in the respective rooms of the reactor building, the turbine building and outside structures.

MCR and ECR are located in the clean section of the reactor building. This section could be affected by radiation during large releases of fission products from the containment. However, MCR and ECR are equipped with filtered air-conditioning systems.

The radiation situation in the MCR and ECR after containment bottom melt-through has not been analyzed yet. The habitability or non-habitability of MCR and ECR would be, in case of such a serious development of a severe accident, evaluated on the basis of measurement of the radiation situation. If MCR and ECR became non-habitable, the staff personnel would have to be evacuated to a shelter and any potential local interventions (if permitted by the current radiation situation) would be organized as intervention groups with pre-determined measures in terms of radiation protection.

In case of both MCR and ECR evacuation, the situation could be solved by design means. There are workstations in TSC; as a standard, they are configured only as an information system and control interventions cannot be performed from them. However, with a change in the configuration of these workstations and removing of staff personnel to TSC, it is possible in this case to secure the implementation of SAMG strategies using the components of non-safety systems to a limited extent.

6.4.4 Conclusion on the adequacy of measures to restrict the radioactive releases

Even though preventing a loss of containment integrity, as the last barrier against fission products releases to the surroundings, is the main goal of SAMG, SAMG also describes strategies to end or reduce the fission products releases after a loss of containment integrity, with the use of all available means.

See the following table for areas for in-depth protection improvement in case of events which may result in a severe accident. The table also identifies areas which need to be subject to additional analysis, since they were not available during the evaluation.

Some measures (identified as "PSR finding" in the note) would be implemented regardless of the targeted evaluation, which confirmed efficiency and correctness of previously adopted decision to implement the particular measures for improved resistance of the original design. For detail description of the corrective measures see the Attachment.

Areas for improvement	**Corrective Action**	**Term** (short-term I / middle-term II)	**Note**
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Analysis	Melt localization out of reactor pressure vessel	II	PSR finding With other VVER

7 General conclusion

7.1 Key provisions enhancing robustness (already implemented)

The objective of safety assessment in the previous chapters was to evaluate the level of robustness and sufficiency of safety margins at Temelin NPP during exposure to extreme natural conditions (considering the facts of the accident at Fukushima NPP), loss of power, loss of ultimate heat sink, and if the event has escalated into a severe accident. A detailed deterministic evaluation was performed to identify the level of defense-in-depth and the capability to fulfill the fundamental safety functions during the specific initiating events and even beyond design basis events regardless of extremely low probability of their occurrence. The evaluation was performed for all reactor operating modes and states, including the case if both Temelin NPP units were affected.

7.1.1 Robustness against earthquakes

No tectonic structures are found in the territory of the Czech Republic that could produce heavy earthquakes. The earthquake level on the site of Temelin NPP shall not exceed, with a 95 % probability, 6.5°MSK-64 (PGA_{hor} = 0.08 g). The buildings, systems and structures important to perform the safety functions can withstand the level of 7°MSK-64 (PGA_{hor} = 0.1 g) at minimum, thus a safety margin is available for the remaining 5 % uncertainty.

The risk of seismic event was analyzed for Temelin NPP together with seismic resistance analysis of buildings, structures and the classified equipment of Temelin NPP. The results of the seismic resistance analysis of buildings, structures and the classified equipment of Temelin NPP show that resistance of all safety related equipment as well as civil structures and buildings where the equipment is located is much higher than the value PGA_{hor} = 0.1 g defined for the maximum design earthquake. There are individual differences in resistance of buildings, systems and structures, but they contribute to further enhancement of safety margins of the safety functions.

The Temelin NPP units are provided with a seismic monitoring system. There are action levels set for the individual earthquake levels with respect to the occurrence of an extraordinary event and emergency response actuation. The NPP personnel are well qualified and trained to assess the damage caused to civil structures and components following a seismic event.

Historical data evaluation and long-term monitoring indicate that the Temelin NPP site is seismically very calm. The results of the detailed seismic zoning network also support the validity of the overall seismic evaluation of the Temelin site and sufficient resistance and margins against consequences of both design and beyond design earthquakes.

7.1.2 Robustness against floods

The Temelin NPP site has never been, nor it may currently be affected by water floods. The main buildings of Temelin NPP where the nuclear safety related equipment is installed are at the elevation of 507.30 m above sea level, which is 135 m above water level of the Hněvkovice dam on the river Vltava. The safety assessment is provided for Temelin NPP in terms of potential dam failure of water storage reservoirs on the upper reaches of Vltava (Lipno I on the river Vltava and Římov on the river Malše). In case of Lipno I dam failure, the Hněvkovice profile would experience app. 10 000-year flood.

During a 10,000-year flood, the Hněvkovice profile would reach a level where most of the pump station used for charging of raw water to Temelin NPP is flooded, thus it would prevent standard operations of raw water supply to Temelin NPP and both NPP units would have to be shut down. The site, however, is provided with sufficient storage volumes to cool down reactors to cold shutdown conditions. During the so-far largest flood on the river Vltava in 2002, the Hněvkovice profile reached a level equal to the maximum elevation considered for this water dam. Water diversion over the dam of the waterworks was done in a standard way and no significant damage was identified either on the pump station for Temelin NPP, or on the waterworks.

Flooding of buildings and structures important to safety by the gravity storm-water drainage system during its regular maintenance is not possible even during extreme precipitation occurrence. In terms of water drainage, Temelin NPP is built up in a cascade where the nuclear safety related buildings and structures are placed at the highest elevation and descending to the border line of the site, which allows natural gravity drainage even in case of failure of the storm-water drainage system. The civil structures of Temelin NPP are designed to withstand floods under the one-day maximum rainfall where the maximum level of 88.1 mm of 10,000-year precipitation is reached in the event of complete failure of the drainage system.

The stream flooding is inherently excluded and Temelin NPP buildings are designed to be resistant against flooding even in rainfall extremes. In addition, the corporate fire brigade has mobile equipment available on the site, which is adjusted to drain local flood water above the maximum 10,000-year values.

7.1.3 Robustness against other weather conditions

The natural phenomena loading is based on statistical processing of data series during at least 30 years of measurement of such events in the Temelin NPP area, or in a similar landscape area. For a design load of weather effects, the recurrence interval is considered 100 years. For an extreme design load of weather effects, the recurrence interval is considered 10,000 years. Buildings and structures of seismic category 1 must withstand the extreme design load effect so that the function of nuclear safety related systems is not compromised. Other buildings and structures are loaded at the design level. The actual resistance levels of seismic category 1 buildings and structures are higher than the resistance values calculated for extreme loading.

The following table shows the specific derived extremes of weather effects on the Temelin NPP site for design levels and extreme design loads (except of precipitation):

Event (Weather Effect)	Design Level (100-year recurrence)	Extreme Design Load (10,000-year recurrence)
Extreme wind [speed]	49 m/s	68 m/s [1)]
Snow [water column equivalent]	92 mm	157 mm
Maximum temperature [immediate value]	39,0 ℃	45,6 ℃
Minimum temperature [immediate value]	-32,3 ℃	-45,9 ℃

1) Including F2 level tornados

Temelín NPP is not threaten by any extreme climatic conditions

7.1.4 Robustness against loss of power

The power supply sources at Temelin NPP provide a sufficient design robustness as well as safety assurance level in the event of loss of off-site power. They are designed with a high level of mutual independence of working and standby house load sources, and further with redundant essential power supply systems that supply safety related systems and components and dispose of their own emergency power sources (DG and accumulator batteries). The house load power supply is designed per each unit to prevent on-site propagation of electrical faults between the units.

Higher design resistance against loss of power is ensured during power operation of the reactor (sufficient defense-in-depth barriers) than during refueling outages. The worst scenario in terms of safety assurance is loss of electrical power to both units at the same time.

There are a total of 8 emergency alternate power sources available on the site (3 safety DGs per unit and 2 common DGs for both units). In the mode of loss of off-site power, Temelin NPP reactors can be maintained in long-term fail-safe state, or cooled down to a cold shutdown state, or safely maintained in the outage mode (power supply is provided to all the necessary mechanical systems and I&C systems) if at least one of the DGs has started up in each unit. For each DG, diesel oil inventory is provided for more than 2 to 3 days, without a need for external refueling. On-site additional diesel oil inventory is available to further extend the DG operation.

The period until discharge of safety systems batteries without charging is several hours depending on the load. Significant prolongation of the discharging period can be achieved by controlled reduction of accumulator load by gradual deployment of individual divisions and use of safety related systems batteries, which have a high capacity. Alternatively other AC sources, available on the site, may be used for long-term charging of batteries, although possibility to connect these sources to the existing power supply system is not discussed or tested in the design.

A robust defense-in-depth exists against total loss of power supply. In spite of that areas for improvement of resistance against SBO have been identified for cases of concurrent failure of all defense-in-depth levels of electric power supply.

7.1.5 Robustness against loss of ultimate heat sink (UHS)

Ultimate heat sink for reactors at Temelin NPP is provided by ambient atmosphere. Ultimate sink for unused heat during power operation of the reactor, or for decay heat after reactor shutdown, is provided in several ways. Heat transfer between heat sources and atmosphere is ensured by the ECW (essential cooling water) system through a spray cooling water tank (ECW cooling pool).

Water inventory at Temelin is available in the spray cooling water tank which can last for about 30 days of operation of ECW system, ensuring decay heat removal from shutdown reactors without external water replenishment to ECW system. A total of 6 ECW pumps are available per each unit. In view of spatial separation of systems and pumps and standalone electrical power supply and other independent support systems, parallel unavailability of all

ECW pumps is extremely unlikely. The fundamental safety functions can even be fulfilled with one pump in operation in one division of ECW system.

A robust defense-in-depth system exists against loss of heat removal to the ultimate heat sink. In spite of that areas for improvement of resistance against loss of UHS have been identified for cases of concurrent failure of all defense-in-depth levels of UHS.

7.1.6 Severe accident management system robustness

Temelin NPP has implemented a severe accident management system to ensure level 4 of defense-in-depth and an emergency preparedness system to ensure level 5 defense of defense-in-depth depth. The functioning and interconnected accident management and emergency preparedness system at Temelin NPP is based on a robust set of personnel, administrative and technical measures. For the personnel area, it is the existence of an emergency response organization and activities related to the individual functions. For the administrative area, it is the implementation of relevant procedures, guidelines and instructions using the capacities of support centers, and for the technical area, it is the functionality of the required scope of technical means for implementation of strategies. Interventions (actions) during the first (preventive) stage of an abnormal event are taken by the personnel of continuous shift operations. Stage two (mitigation of consequences) begins if the scope of event is getting beyond the capacity of the shift personnel where the emergency response organization is activated. In this case, the responsibility for management of interventions (actions) is taken over by the emergency staff of Temelin NPP with the assistance of the technical support centre.

In case of an abnormal event, all the necessary activities would be managed and performed from places protected against the effects of activity leak into the environment. The technical support centre and emergency staff who are responsible for management of strategies according to SAMG are based in the emergency control center. It is a protected workstation enabling permanent occupancy for a period of at least 72 hours without external support. Remote activities to implement strategies would be performed by the shift personnel from MCR or ECR. Local activities and possible equipment repairs in the relevant cold area of the reactor building, turbine hall or support buildings would be provided by intervention teams based in the operating support center.

The concept of the accident management system at Temelin NPP is based on symptomatic approach. Strategies are developed for the management of technological accidents (up to fuel failure) and included in EOPs, with the highest priority to recover heat removal from the core and to prevent damage on the first barrier against leak of fission products (fuel cladding). Strategies are developed for mitigation of consequences of fuel failure accidents (severe accidents) and included in SAMG, with the highest priority to prevent damage on the third barrier against leak of fission products (containment) which at that time is the last intact barrier. EOPs and SAMG are updated on a regular basis to incorporate knowledge from training of their use on the simulator, or emergency exercises as well as external knowledge.

The emergency preparedness system is implemented in case of safety challenge to the reactor (unit) or site area, or under circumstances which the shift personnel are unable to manage. The emergency response organization (OHO) would be activated in case of any level of abnormal event (Alert, Site emergency, General emergency). Its internal part (IOHO) consists of shift personnel and its emergency part (POHO) consists of NPP technical staff experts who are on call duty.

A system of qualification requirements has been implemented to select shift personnel and experts for POHO and additional criteria are considered to take their knowledge and expertise into account. The readiness of shift and technical personnel to manage technological accidents is tested on a periodical basis during full-scope simulator training with the participation of the Technical Support Center staff and during emergency exercises.

The organization of abnormal event (emergency) management (including severe accidents) is defined in the on-site emergency plan approved by SONS (State Office for Nuclear Safety).

If emergency conditions occur (design basis or beyond design basis event without fuel failure), all currently available technical means within their scope of design specification will be employed to meet the EOP requirements. SAMG envisage performing the required actions using all available systems and components, or all available technical means even outside design specifications.

There is a corporate fire brigade unit available on the Temelin site which disposes of the relevant fire protection equipment and they are trained to intervene at any point of the site area. The pumping equipment of the fire brigade belongs to the major mobile non-technological means available for transport and pumping of fluids.

The accident management program at Temelin NPP is analytically supported for a long term. Analytical support is based on probabilistic and deterministic approach, consisting in selection of the most likely emergency scenarios leading to severe accidents, and consequently in their deterministic analysis using integral computation codes. The result of analytical support is a summary of knowledge based on understanding of phenomena during severe accidents and their timing, identification of possible project weaknesses, and determination of activities to mitigate consequences of severe accidents, and validation of severe accident response activities, and determination of the source term to evaluate possible radiological consequences. Analytical support will also produce a simulation tool to display phenomena during scenarios of severe accidents.

The accident management and emergency preparedness system contains robust set of personnel, administrative and technical measures. In spite of that areas for improvement of efficiency of these measures have been identified for management of beyond design basis, highly improbable events.

7.2 Safety issues

The assessment of safety margins of Temelin NPP in extreme natural conditions, loss of electrical power, loss of ultimate heat sink, and the capability to manage a situation where the event is escalating into a severe accident confirmed for the majority of emergency scenarios that sufficient margins exist and barriers are robust enough to provide defense in depth both in the area of design and in the area of personnel, administrative and technical provisions for accident management.

In spite of considerable robustness of barriers, it was concluded based on results of assessment of safety margins for initiating events, loss of safety functions, and provisions for beyond design basis and severe accident management at Temelin NPP that opportunities for further safety enhancement exist with respect to highly improbable beyond design basis situations.

For each identified potential, the relevance was defined in terms of the size of safety margins, i.e. resistance against possible loss of the capability to fulfill the fundamental safety functions and preparedness to manage any occurring situations. In assessing the risk relevance, we took into account the number of in-depth defense barriers that would have to fail before given conditions may occur and the time for which the unit is able to withstand an event with the existing safety margins. By that time, it is necessary to have sufficient means available to fulfill the required functions, or take consequent protective actions to reduce personal exposure for protection of people.

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Attachment

Description of proposed corrective measures

The attachment includes short description of proposed corrective measures for all areas evaluated within the Stress Tests. The following information is provided for each corrective measure:

Function

Safety function improved by the said corrective measure.

Solution method

Short description of the method of solution including proposed options.

Risks

Risks managed by implementation of the proposed corrective measure.

Next step

Short description of the method of implementation of the corrective measure.

	FUNCTION	CORRECTIVE ACTION - SOLUTION METHOD	RISKS	NEXT STEP
ALTERNATIVE TECHNICAL MEANS	**1. Heat removal from core through RCS**	Alternative method of water make up to (a) SG/SFP from TX tanks (with possible secondary heat removal from the RCS), (b) RCS/SFP (with open RCS) from tanks containing H_3BO_3 (TB30) using: i) stable motor driven pump outside unit (max. power app. 140 kW) (one per unit, mobile) ii) fire truck Establishment of connection points from appropriate distribution lines (flanges, penetrations, valves, option to install "flexible" connections) to the existing pipe routes (TX10, TQ, TG systems). Possibility of alternative source of water after depletion of specified tanks.	Flooding UHS SBO	These are diverse (independent) means connected to electric power distribution lines / piping routes of existing safety systems, i.e. solution affecting the design of existing BS. IMMEDIATELY: coordination with other VVER operators / WANO / VVER designer – clarification of the concept 2012: feasibility study
	2. Heat removal from core through secondary circuit		Flooding UHS SBO	
	3. Pumping of medium for SFP (including SFP make-up)		Flooding UHS SBO	
	4. Provision of category I power supply	DG 400 kW (0.4 kV / 50 Hz) to provide voltage for category I ZN + voltage for selected 0.4 kV category II loads. (HVAC for MCR/ECR), selected valves, or possibly also fire extinguishing system). (Container DG (one per unit, mobile), connection by cable line to selected (any) train (0.4 kV Cx03 + fuel feeding) Note: for I&C cooling: Shutdown of selected cabinets, open the doors ... no special means are probably necessary	SBO	
	5. Habitability of the MCR/ECR		SBO	
	6. Provision of heat removal from I&C systems		SBO UHS	

	FUNCTION	CORRECTIVE ACTION - SOLUTION METHOD	RISKS	NEXT STEP
TECHNICAL MEANS - DESIGN BASED	Long-term availability of emergency systems of category II sources (DG) under any extreme conditions	Improvement of DG building resistance against external flooding - ZKZ D041 (addition of sealing, barriers, insulation coverage). (The risk of flooding of 100m^3 tanks and diesel fuel pumps pumping fuel to the operating fuel tank, etc. …)	Earthquake Flooding Extreme situations LOOP	Implementation of ZKZ D041
		Establishment of a connection to the storage tanks and transport by a truck tank with own pump to DG fuel tank - provided in PP 0TS276/2 - common part. The truck tank for diesel fuel supply should be provided externally. *(Needed after app. 70 hours of DG operation ...)*		
	Provision of CNTN isolation	Power driven CNTN isolation valves supplied from category II ZN - instruction for manual closing or switch over to category I (related to TL22/42 systems directly connected with containment atmosphere).	SBO	
	Plant control in case of ECR flooding (internal flooding).	Measures preventing water penetration to ECR - modification of entrance doors to ECR (sealing). *Note: ECR flooding after rupture of ECW piping will not result in false generation of commands and negative influence of I&C (… architecture, control coding).*		ZKZ initiation
	Coolant inventory inside CNTN available for ECCS	i) Standard make up of sump from TM - installation / replacement of pumps (TM) - higher delivery capacity. ii) Alternative make up of sump from TB30 or TB10 tanks by newly installed pumps and pipe routes.	AMP	Conceptual design, ZKZ initiation
	Infrastructure - possibility of intervention, access to SO, ...	i) Verification of resistance of selected buildings against seismic event - SO 656/01 (fire brigade station), SO 643/01 (service gas management), SO 410/03 (boiler house), AB, bridges, covers ii) Analysis of access to selected buildings	Earthquake	Analysis

	FUNCTION	CORRECTIVE ACTION - SOLUTION METHOD	RISKS	NEXT STEP
TECHNICAL MEANS - SA	Hydrogen disposal system	Hydrogen recombiners - add PAR to quantity that will be able to manage beyond design basis accidents - ZKZ B462	AMP	Implementation of B462
	Analysis of systems for use in SAMG	Verification of system functions in beyond design basis operating conditions in accordance with SAMG (review of specified SAMG operations from plant point of view - control options, parameters, etc.)	AMP	Analysis
	Melt localization out of reactor pressure vessel	Perform analysis of door opening and installation of barriers for melt localization.	AMP	----
	Provide habitability of the MCR/ECR after SA transition to ex-vessel phase	Analysis of radiation situation in MCR/ECR after melt penetration from RPV to CNTN and CNTN damage, analysis of radiation situation in case of radiation leakage through HVAC.	AMP	

	FUNCTION	CORRECTIVE ACTION - SOLUTION METHOD	RISKS	NEXT STEP
PROCEDURES / GUIDELINES	Qualified alternative source for SBO management of one unit - provision of voltage from DG of the other unit to one of the train switchyard BV, BW, BX.	Implementation of procedure to existing instructions for SBO - alternative with system DG of the other unit as the alternative source.	SBO	2012: Provide analysis, complete instruction
	Prolongation of discharging period of accumulators of category I ZN during SBO	Analysis of accumulator discharging period with application of controlled reduction of load (switch off of loads). Completion of instruction for SBO - instructions for reduction of category I ZN load	SBO	
	Completion of operating documentation - long-term power supply from DG	Completion of instruction	LOOP	
	CNTN closing	Measures for early closing of CNTN in case of accidents during outage - completion of instruction *(Solution within SOER 21010-1, Safety during outage)*	SBO UHS	
	SOER for outage / SA in SFP	Develop „shutdown SAMG “.	AMP	2012: Initiation
	Instructions/EDMG for alternative means	Develop EDMG type instructions	AMP	In relation of alternative technical means solution
	SFP cooling	Analysis/calculation of the period of heat removal from SFP without make up - for use in instructions (development of a tool for TSC - heat output in SFP versus period until fuel assemblies exposure).	SBO	Analysis

	Function	Corrective action - solution method	Risks	Next step
Personnel	Emergency organization staff, personnel training	Emergency organization staffing criteria (highest qualification personnel) - verification. Review/define concept of personnel training in SA area.	AMP	2012: Solution /implementation of measures.
	Provision of enough personnel	i) Measures for rotation of shift personnel in case of aggravate accessibility of the location. ii) Analysis of possibilities of shift personnel in case of an accident on both units (enough staff for implementation of strategies, sheltering of personnel). iii) (After EDMG development - analysis of conditions and options / enough staff for implementation of actions in accordance with EDMG) *Note: Quick evacuation of plant personnel (personnel not involved in accident management) in case of unavailability of shelters - EP measures.*	Earthquake Flooding Extreme situations SBO AMP	
	Emergency control centre usability - ability of emergency organization staff to operate out of emergency control centre	Activation of TSC and emergency commission out of emergency control centre (České Budějovice, MCR, ECR, including provision of information and communication) - verification of possibilities, incorporation into documentation	Earthquake Flooding	

	FUNCTION	CORRECTIVE ACTION - SOLUTION METHOD	RISKS	NEXT STEP
COMMUNICATION	Provision of long-term communication between intervening personnel and communication with external authorities	<u>Warning and notification in NPP and EPZ</u> Define alternative organizational solution (in cooperation with state authorities and IZS) in case of unavailable sound broadcasting and hooters (or possibly control infrastructures for their activation) because of extreme natural events (provision of mobile means - mechanical hooters, pneumatic hooters, mobile megaphones, mobile hooters) including method of use and incorporation into emergency planning documentation. <u>Communication between intervening personnel and external communication</u> (a) On the site (MCR – TSC – intervening personnel …) i) Telephone exchange or radio communications network - alternative power supply of telephone exchange (AB, ÚED, HVBI,II, BAPP) from mobile portable DG (up to 10 kW). Alternatively the sources may be used also for emergency power supply of radio communications network components (couplers, amplifiers) (as chosen in actual situation) (Note: existing back-up of telephone exchange in case of loss of power supply is provided from accumulators for app. 8 hours, for radio communications network it is about 6 hours) ii) Determine concept for communication in case of communications network decomposition as a result of a seismic event *(all existing communication infrastructure (telephones, radio stations) depends on centralized components installed in non seismic resistant buildings, non seismic resistant cable channel etc. – currently there are redundant communications options available (fixed lines, MT, radio stations), however without full functionality guarantee after PZ/MVZ)* … in cooperation with other operators. (b) Between the power plant and the external authorities See section ii) above. Verification of possibilities of communication at key workplaces (power plant, state authorities) using satellite phones.	Earthquake SBO	2012: Coordination with state authorities and IZS in the area of alternative warning methods, notification and communication in case of damage of existing infrastructure. Modification of emergency planning documentation. Coordination with other operators in the area of communications. Possible analysis of resistance of components in case of DE assumed confirmation of at least limited functionality.