ANNOTATION

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 analysing 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.

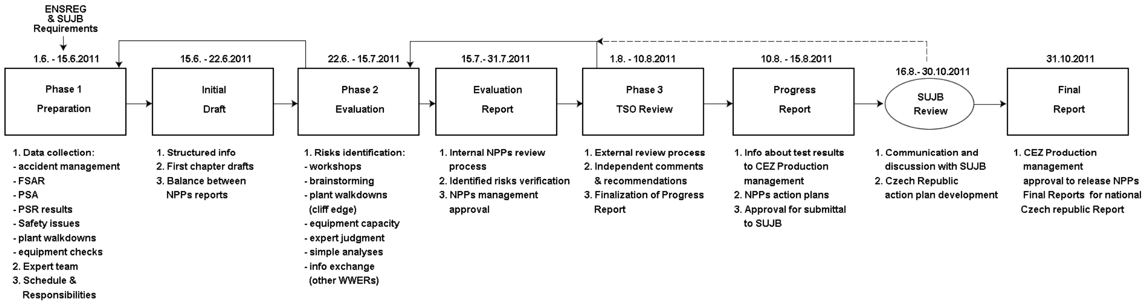
Based on the facts concerning the accident in the Fukushima-Daichi NPP, the international nuclear authorities have issued a number of findings and lessons learned for the nuclear industry, which are applicable to all types of reactors. The submitted report contains outcomes of the stress tests specified by the ENSREG´s (European Nuclear Safety Regulators Group) declaration of 13 March 2011 "EU Stress Tests Specifications". The stress tests are part of the comprehensive review of the NPP's safety linked to the international documents published about the respective 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 21st century, 12 July 2011

ČEZ, a.s. was requested to perform the stress tests in the letter of SÚJB of 25 May 2011. Performance of the stress tests was regulated by the order issued by the manager of the Production Division ČEZ, a.s. who specified scope and method of implementation thereof.

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

Considering the short time limit within which the stress tests had to be performed, a working team was appointed, a fixed time schedule was determined and outcomes of individual stages of processing thereof were defined. On 15 August, SÚJB was informed on the review´s current state and progress in the form of the so called Progress Report.



In the course of processing of the review, multiple working meetings were held with the authors of the stress tests for the other NPPs of the VVER type within the framework of the so-called VVER club (EDU, Paks, Loviisa, Bohunice, Mochovce) and the Kozloduy NPP. Within the framework of WANO MC, discussions are taking place with other operators of NPPs of the VVER from non-EU countries.

An independent review of the outcomes by the most important external suppliers in the field of nuclear safety, among others in particular ÚJV Řež and Westinghouse, has been performed to provide for impartiality thereof.

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

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 in-depth defence 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 in-depth defence.

The assessment covers all operating modes and states of nuclear units. 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 EDU 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;

• Possibility to use Fire Brigade pumps for cooling or make-up

• Location with minimum seismic risk;

• Location practically eliminating external floods;

• Two large storage reservoirs to replenish raw water;

• Extensive supply of cooling water on the site;

• Compact spent nuclear fuel pools ensuring fuel sub-criticality even if flooded with clean water

• Large content of hermetic areas (bubbler system) and relatively lesser source term (lower output parameters of the reactor); and

• Large volume of primary coolant water inside the containment (water inventory in bubbler system)

Safety of the Dukovany NPP was confirmed in the past by a number of international missions assessing the level of nuclear safety security considering experience of the western NPPs. In particular, the following missions were concerned:

- Operational Safety Review (OSART), IAEA (1989),
- Operational Safety Review Follow-up (OSART FU), IAEA (1991),
- Assessment of Significant Safety Events (ASSET), IAEA (1993),
- Assessment of Significant Safety Events (ASSET), IAEA (1996),
- Technical Audit, ENAC (1994 – 1995): review of technical condition of the EDU systems (a base of the Equipment Refurbishment Program),

- Safety Issues of WWER 440 Resolution Review, IAEA (1995): review of safety findings concerning material “Safety issues and their ranking for WWER 440 model 213 Nuclear Power Plants”,
- Nuclear Insurance Pool, Marsh & McLennan (1996): fire risks assessment,
- Nuclear Insurance Pool, organized by Gradmann & Holler (1996): assessment of machinery failure risk,
- Nuclear Insurance Pool, ČJP (1997): insurance risks inspection (in December 1997, the insurance contract covering liability risks was signed between ČEZ, a. s., and ČJP),
- Peer Review, WANO (1997): inspection of the systems and working procedures according to the INPO criteria,
- International Peer Review Service (IPERS), IAEA (1998): review of the probabilistic safety model of the NPP and determination of the core meltdown probability,
- International Physical Protection Advisory Service (IPPAS), IAEA (1998): review of physical protection of the nuclear facilities and protection of the nuclear materials and the nuclear facilities,
- Nuclear Insurance Pool, ČJP (2000): international inspection of insurance risks, verification of the set conditions´ maintenance,
- EMS Certification Audit, Det Norste Veritas (2001): final review of the Dukovany NPP according to requirements imposed by standard ISO 14001 - Eco-friendly Undertaking, subject to regular recertification every 3 years
- Operational Safety Review (OSART), IAEA (2001): review of operational safety concerning all fields of the NPP´s operation,
- Operational Safety Review Follow-up, (OSART FU), IAEA (2003): subsequent mission concerning fulfilment of the recommendations of 2001,
- Nuclear Insurance Pool, ČJP (2006): inspection of fulfilment of recommendations resulting from the previous inspection,
- Peer Review, WANO (2007): inspection concerning the NPP’s operational safety areas by the operators of the other nuclear power plants,
- Safety Assessment Long Term Operation Review (SALTO), IAEA (2008): review of the NPP´s readiness to secure safety aspects resulting from the NPP´s service life´s extension,

- Peer Review Follow-up, WANO (2009): subsequent mission to inspect the implemented correction actions concerning the mission in 2007,
- Operational Safety Review (OSART), IAEA (2011): re-review of the NPP´s operational safety by international specialists,

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 respected while selecting a site for Dukovany NPP. This regulation specified siting of future NPPs in view of minimizing risk relating to external effects (now included in the regulation of the Czech regulator – State Office for Nuclear Safety (SÚJB) no. 215 of 1997).

For that reason, the site of EDU is considered as highly stable in relation to seismicity. The end process medium (ultimate heat sink) is atmosphere and cooling is provided by evaporation from cooling towers. 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 estimate is used. The report analyses 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).

In spite of the Dukovany NPP´s high level of operational safety and robustness, the following possibilities were defined on the basis of the executed analyses to increase resistance of EDU 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 organisation and staff training for management of extreme conditions (e.g. when both NPP units are affected, or in case of loss of control centres, communication systems, etc.)

- Extend the capacity of recombiners to eliminate undesirable consequences of excessive hydrogen production during severe accidents and increase the resistance containment.

In 2006 and 2007, the comprehensive safety review of the Dukovany NPP after 20 years of operation was performed (the so-called Periodic safety review for all 14 areas according to the IAEA NS-G-2.10 guideline). The review concerned in-depth inspection of fulfilment of requirements imposed by the national as well as international legislative documents, the WENRA requirements defined in the “Reactor Safety Reference Levels” documents and other international recommendations arising from the IAEA (Safety Guides, INSAG) documents. Actually, the comprehensive review within the PSR´s framework has identified similar opportunities to increase safety as those stated in this report. Some of them (improvement of the EDU´s design´s resistance to consequences of severe accidents comprising increase of the hydrogen disposal system´s capacity, completion or recombiners for severe accidents and preparation for the reactor cavity´s flooding) are today in the stage of preparation for implementation and would be implemented regardless of this new review. In fact, PSR assumes the implementation of the approved actions within 2015, i.e. within the time limit for the Dukovany NPP´s service life´s extension.

The outcomes of the goal-directed review of safety margins and resistance of the NPP requested by the European Council confirm effectiveness and accuracy of the decisions taken previously with a view to implementing measures aiming at upgrade of the original design. No conditions were identified that would require an immediate solution. The power plant is able to safely manage even highly improbable extreme emergency conditions without posing any threat to its vicinity.

List of Abbreviations:

AFWP	Auxiliary feed water pump
AOPs	Abnormal Operating Procedures
ATWS	Anticipated Transient Without Scram
BVP	Reactor refueling pool
CCHV	Cooling Water
CČS	Central Pumping Station
CDF	Core Damage Frequency
CHV	Cooling Tower
CISRK	Radiological Central Information System
ČSJ	Jihlava River Raw Water Pumping Station
DG	Diesel Generator
EBO	Bohunice NPP
EC	Emergency Commission
ECC	Emergency Control Centre (EC + TSC)
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
EDA	Dalešice Hydro-Plant
EDU	Dukovany NPP
EFWP	Ultimate Feed Water Pump
ELS	Emergency Load Sequencer
EMO	Mochovce NPP
EOPs	Emergency Operating Procedures
EPZ	Emergency planning zone
ESW	Essential Service Water
ETE	Temelín NPP
FB	On-site Fire Rescue Brigade
FDF	Fuel Damage Frequency
FWT	Feed Water Tank
HVB	Main production building (reactor and turbine hall for two units)
I.O	Primary Circuit
IAEA	International Atomic Energy Agency
II.O	Secondary Circuit
IZS	Integrated Rescue System
LERF	Large Early Release Frequency
LOCA	Loss Of Coolant Accident
LOOP	Loss of Offsite Power
MCR	Main Control Room
MU	Emergency situation
NPP	Nuclear Power Plant
NTKS	Low Pressure Compressor Station
OHO	Emergency Response Organization

PAMS	Post Accident Monitoring System
PAR	Passive Recombiner
PGA	Peak ground acceleration
POHO	Permanent Emergency Response Organization
PORV PRZR	Power Operated Relieve Valve PRZR
PRZR	Pressurizer
PSA	Probabilistic Safety Assessment
QOV	Quick Operating Valve
RCP	Reactor Coolant Pump
RHRP	Residual Heat Removal Pump
RMMS	Mobile fast monitoring group
RNVS	Stand-by power supply for own consumption
RPV	Reactor Pressure Vessel
RSD	RHR Reduction Station
RTARC	Real Time Accident Release Consequence – TSC SW programme
RTS (HO-1)	Reactor Trip System
SAG	Severe Accident Guidance
SAMG	Severe Accident Management Guideline
SBO	Station Black Out
SCG	Severe Challenge Guideline
SDEOPs	Shut Down EOPs
SEJVAL	Radiological control System of discharges
SFP	Spent Fuel Pool
SG	Steam Generator
SGSV	Steam Generator Safety Valve
SMRK	Head Master of Radiological protection
ŠP	Fission product
SSC	System, Structure, Component
SUJB	State Office for Nuclear Safety
SV PRZR	Pressuriser Safety Valve
SYRAD	Radiological Information System
TDS	Radiological monitoring system on the border of EDU
TG	Turbinegenerator
TG	SFP cooling system
TK	RHR condenser
TSC	Technical Support Centre
TSFO	Security Technical Systems
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply (Source)
VE	Hydro-Plant
VRV	Vranov Hydro-Plant
VTKS	High pressure Compressor station
VVN	High Voltage

AFWP	Auxiliary feed water pump
AOPs	Abnormal Operating Procedures
ATWS	Anticipated Transient Without Scram
BVP	Reactor refuelling pool
CCHV	Cooling Water
CČS	Central Pumping Station
CDF	Core Damage Frequency
CHV	Cooling Tower
CISRK	Radiological Central Information System
ČSJ	Jihlava River Raw Water Pumping Station
DG	Diesel Generator
EBO	Bohunice NPP
EC	Emergency Commission
ECC	Emergency Control Centre (EC + TSC)
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
EDA	Dalešice Hydro-Plant
EDU	Dukovany NPP
EFWP	Ultimate Feed Water Pump

1 General data about the sites and nuclear power plants

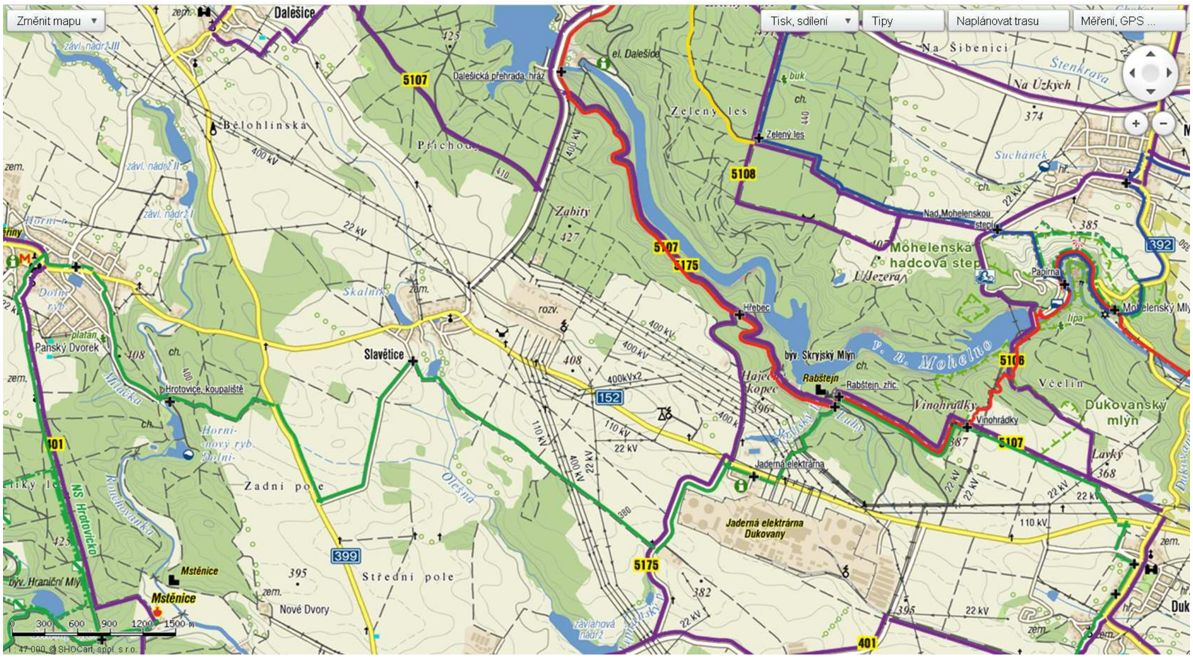
1.1 Brief description of the sites characteristics

1.1.1 Location characteristic

The Dukovany nuclear power plant (EDU) is situated to the southwest of Brno on the selected flat surface at the altitude of 389.3 m above sea level which is confined in the north by the deeply incised Jihlava river´s valley. The nearest settlements and the municipalities are Mohelno, Dukovany, Rouchovany and Slavětice, all within a distance of 3 to 5 km. To the northeast of the NPP premises, the Moravské Budějovice - Brno road communication of class 2 No. 152 is situated. The EDU is connected to the railway line from the eastern direction by means of a rail access from Rakšice connection station on the Moravský Krumlov - Brno ČD railway line.

The subsoil of the individual civil structures of the Dukovany NPP consists of rock material or residual soils thereof and therefore, considering the geotechnical parameters of the aforementioned materials, it is possible to regard the foundation conditions, within the meaning of ČSN 73 1001 Art. 20a, as non-complex.

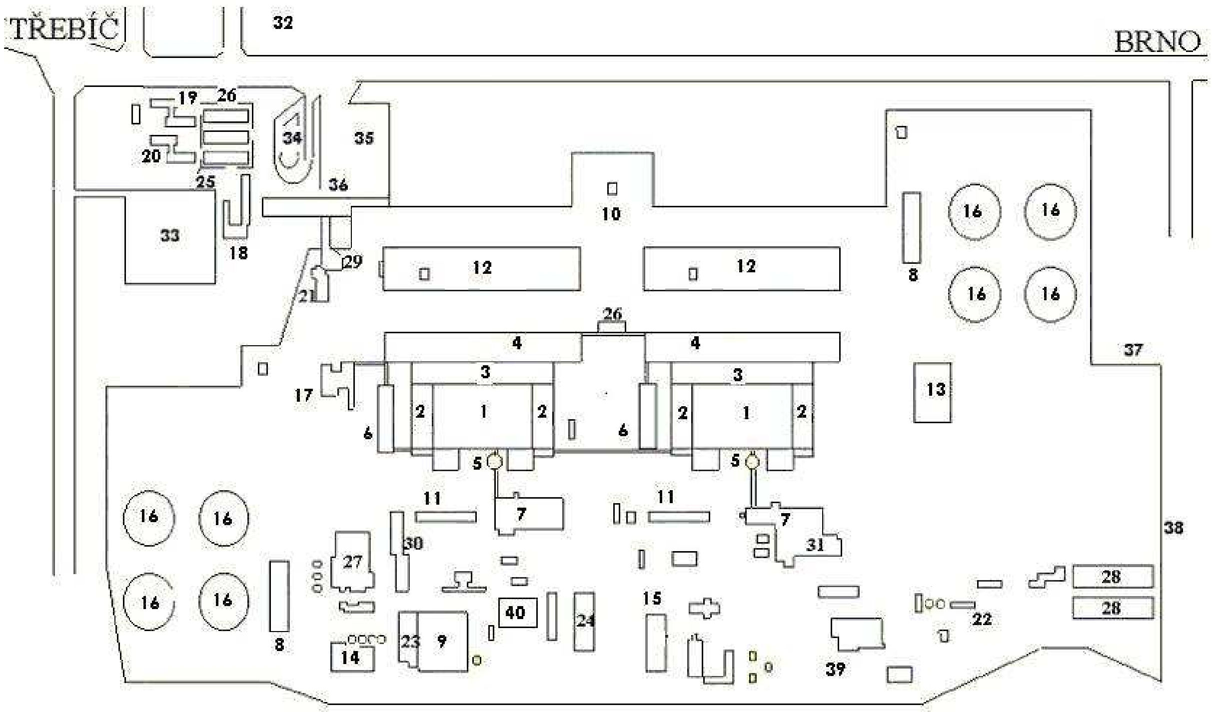
For the foundation of the civil structures having critical importance for safety of the nuclear power plant´s operation, weathered as well as sound rock materials were selected; in places of tectonic disturbances where the rock material decomposed to form soils of the clayed sand type, such sands were extracted and replaced with a filling of structural concrete.



Four nuclear units (using some common equipment) are operated in the EDU location. The units are identical, structurally combined into twin units.

On the Jihlava river, there is Dalešice pumped storage hydroelectric plant which serves as a water reservoir for the nuclear power station as well. Electric power is led out from the nuclear power plant to the 400kV substation Slavětice.

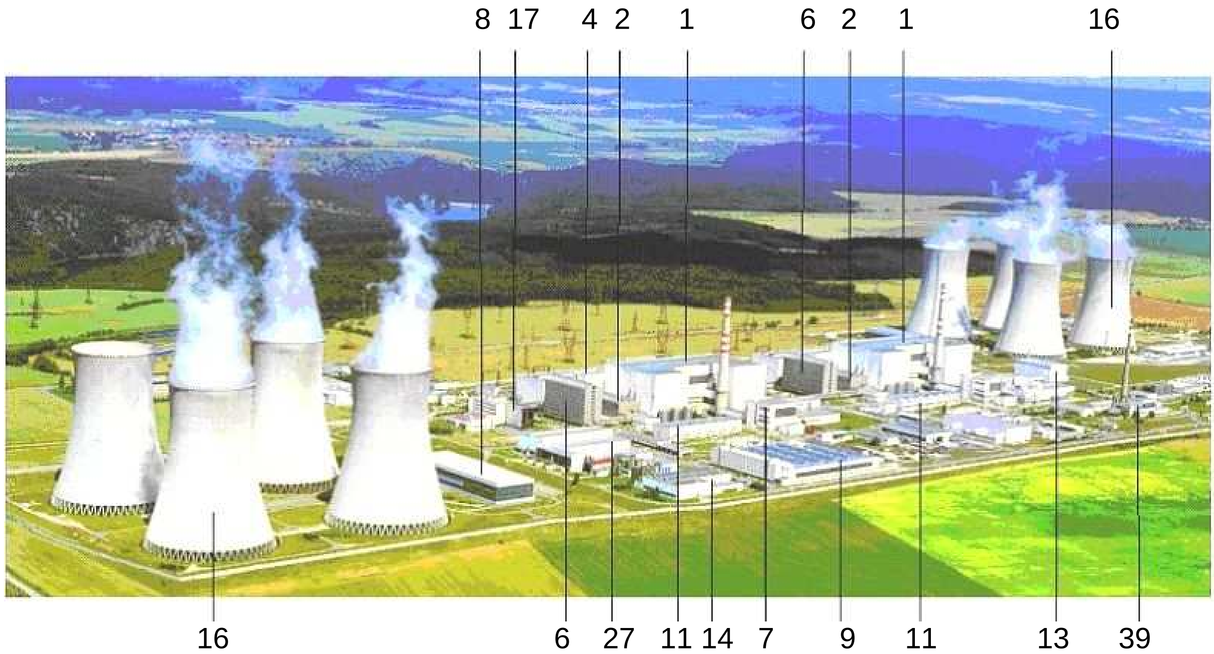
In addition, there are 2 spent nuclear fuel storages (MSVP/SVP) situated in the EDU location. Spent fuel is stored in the CASTOR-type casks cooled by natural air circulation in MSVP/SVP. With respect to the passive principle of the casks´ cooling, there is no risk of loss of the ability to fulfil the safety function in case of occurrence of an initiating event and this means that MSVP/SVP is not the subject of this evaluation of safety and safety margins.



The layout of the important civil structures of the Dukovany NPP:

1-Reactor building, 2-Cross intermediate room, 3-Longitudinal intermediate room, 4-Turbine building, 5-Vent discharge stack,7-Nuclear auxiliaries building, 8-Central pumping station,10-Service water purification system, 11-Dieselgenerator station, 12-Switchyard 400 kV, 13-Interim long-term storage of spent fuel and spent nuclear fuel storage, 14-Chemical water treatment, 15-Compressor plant and cold supply plant, 16-Cooling tower, 17-Administration building 1 (shelter of the Emergency Control Centre (HŘS)), 28-Radioactive waste repository, 29-Gatehouse, 30-On-Site Fire Rescue Brigade, 32-Waste water purification system and two retention basins

Aerial view of the nuclear power plant with identification of certain civil structures



The holder of the licence to operate all nuclear facilities found in the location is ČEZ, a. s., Duhová 2/1444, 140 53 Praha 4. The currently effective permits to operate EDU were issued as follows: for the first unit by Decision of the State Office for Nuclear Safety (SÚJB) ref. No. 24273/2005 dated 16 December 2005, for the second unit by Decision of SÚJB ref. No. 55714/2006 dated 8 December 2006, for the third unit by Decision of SÚJB ref. No. 30852/2007 dated 10 December 2007 and for the fourth unit by Decision of SÚJB ref. No. 30853/2007 dated 10 December 2007. Validity of all the permits expires in 10 years.

1.2 Main characteristics of the units

1.2.1 General unit description

The individual RBs of EDU contains pressurized water reactors VVER 440 (type V-213č) the output of which is either 1,375 MWt or 1,444 MWt. The units were commissioned during 1985 – 1987.

The reactor (or the reactor core) is cooled and moderated by water of the primary circuit which is pumped through the core by means of the main circulation pumps. After passage through the reactor, heat accumulated in the coolant is exchanged with water of the secondary circuit in the steam generators. Pressure in the primary circuit is maintained by the pressurizer. The reactor coolant system (primary circuit) consists of six loops of the main circulation piping (HCP), six horizontal steam generators (PG), six main circulation pumps (HCČ), twelve main closing valves (HUA) and the pressurizer system.

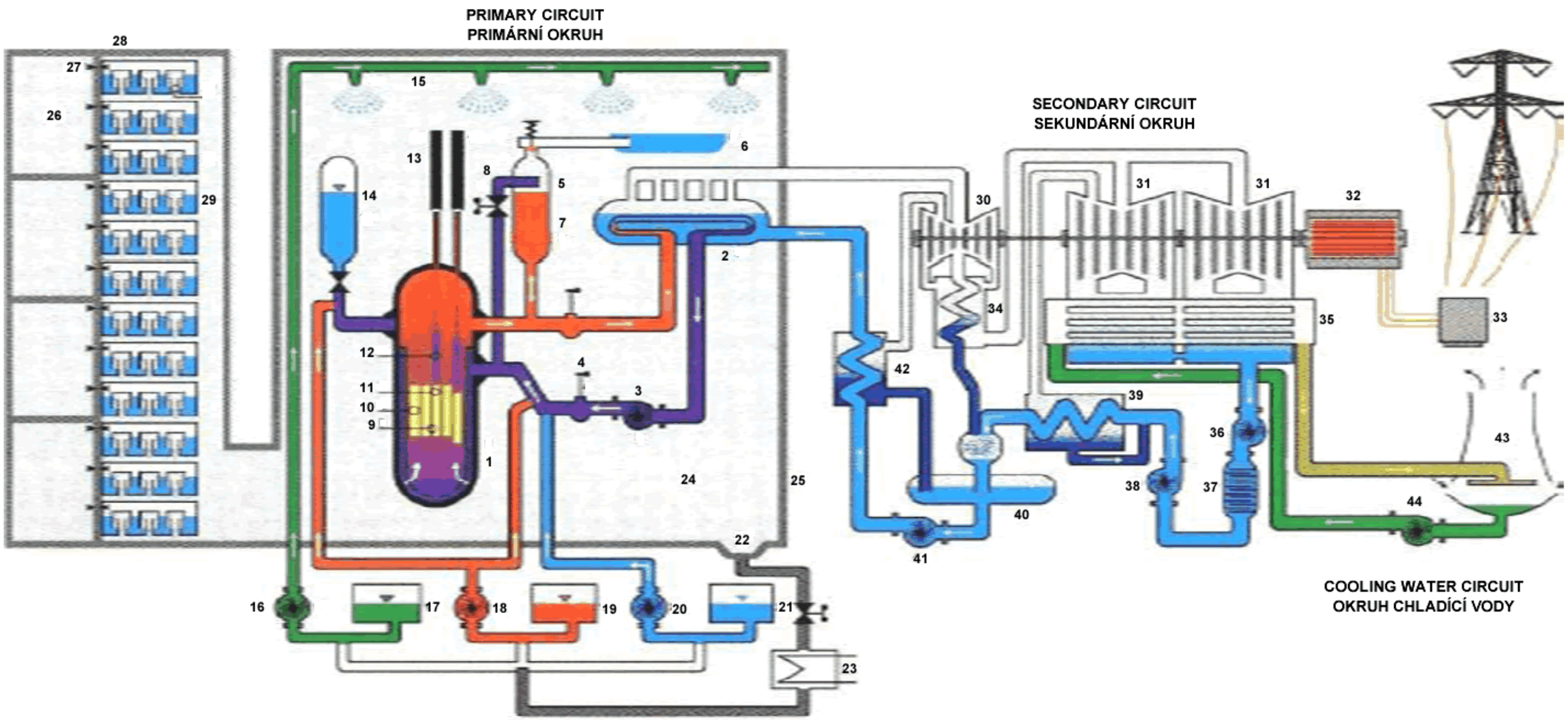
The reactor and the man components of the primary circuit are located in the hermetic zone which forms a reinforced-concrete barrier against release of radioactive media to the environment. The hermetic zone (hereinafter referred to as "the containment") is located inside the reactor building which is situated above the main floor on the level of 18.9 m and continues by the steel structure forming its roofing. In the reactor building, there are spent fuel storage pools (BSVP) where spent fuel from the core is being deposited. From here, after reduction of residual output, the fuel is transported to MSVP/SVP on a continuous basis.

The secondary circuit consists of two turbine generators for one unit, two condensers per turbine and the piping for steam, condensate and feed water. The secondary circuit is followed by the circulation cooling water systems and the service water systems with four cooling towers for HVB which are sometimes called the tertiary system in summary.

Residual heat is removed to the atmosphere under normal operation through the steam generators, main condensers, circulation water and cooling towers, and during outage, through the steam generators, technological condensers, essential service water system and cooling towers. Heat removal from the essential service water system to the atmosphere has been implemented by an independent distribution system of this water in the natural-draught cooling towers. The essential service water pumping station is designed as an independent civil structure for the twin-unit; which means that there are two TVD pumping stations in the nuclear power plant´s premises in total.

The active safety systems have the 3 x 100 % redundancy and are mutually independent and physically separated. The passive safety systems (the hydro accumulators inside the containment) have the 2 x 100 % redundancy. Seismic resistance of all the redundant safety systems, including the power supply and the control systems as well as all the other auxiliary systems is provided. The stand-by power supply units of the power supply systems and the control systems are mutually independent, physically separated and seismic-resistant (they are subject to qualification applicable to safety systems). The design disposes of diversification of the systems for the fulfilment of three basic safety functions - 1) provision of the reactor shutdown (sub-criticality), 2) heat removal (after-cooling) and 3) elimination of releases (barriers and the containment´s isolation) - see chap. 1.3.

Fig. 1.2.1-1: Technological layout of EDU:



1 - Reaktor, 2 - Parogenerátor, 3 - Hlavní cirkulační čerpadlo, 4 - Hlavní uzavírací armatura, 5 - Kompenzátor objemu - pára, 6 - Barbotážní nádrž, 7 - KO - voda, 8 - Vstřiky KO, 9 - Aktivní zóna, 10 - Palivová kazeta, 11 - Regulační kazeta (HRK), palivová část, 12 - Regulační kazeta (HRK), absorbční část, 13 - Pohony HRK, 14 - Hydroakumulátor,15 - Sprchový systém, 16 - Sprchové čerpadlo, 17 - Zásobní nádrž sprchového systému, 18 - Nízkotlaké havarijní čerpadlo, 19 - Zásobní nádrž nízkotlakého havarijního systému, 20 - Vysokotlaké havarijní čerpadlo, 21 - Zásobní nádrž VT havarijního systému, 22 - Sání z hermetické zóny, 23 - Chladič sprchového systému, 24 - Kontejnment, 25 - Ochranná obálka kontejnmentu, 26 - Záchytný plynojem barbotážní věže, 27 - Zpětná klapka, 28 - Brbotážní věž, 29 - Žlaby barbotážní věže, 30 - VT díl turbíny, 31 - NT díl turbíny, 32 - Elektrický generátor, 33 - Blokový transformátor, 34 - Separátor a přihřívač páry, 35 - Kondenzátor, 36 - Kondenzátní čerpadlo I°, 37 - Bloková úprava kondenzátu, 38 - Kondezátní čerpadlo II°, 39 - NT regenerace, 40 - Napájecí nádrž, 41 - Elektronapájecí čerpadlo, 42 - VT regenerace, 43 - Chladící věž cirkulační vody, 44 - Čerpadla CV

(Legend: 1 – reactor, 2 – steam generator,3 – main circulation pump, 4 – loop isolation valve, 5 - pressurizer – steam, 6 – bubbler tank, 7 – KO – water, 8- KO injections, 9 – core, 10 – fuel assembly, 11 – control rod (HRK), follower, 12 – control rod (HRK), blocking part, 13 – HRK drives, 14 – hydroaccumulator, 15 – spray system, 16 – spray pump, 17 – spray system storage tank, 18 – low-pressure safety injection pump, 19 – storage tank of the low-pressure safety injection system, 20 – high-pressure safety injection pump, 21 - storage tank of the high-pressure safety injection system, 22 – suction from the hermetic zone, 23 – spray system cooler, 24 – containment, 25 – primary containment, 26 – gas holders of the bubbler tower, 27 – check flap valve, 28 – bubbler tower, 29 – bubbler tower´s conduits, 30 – high-pressure part of the turbine, 31 – low-pressure part of the turbine, 32 – electric generator, 33 – generator transformer, 34 – separator and steam reheater, 35 – condenser, 36 – condensate pump I*, 37 – condensate demineraliser, 38 – condensate pump II*, 39 – low-pressure regeneration, 40 – feed water tank, 41 – auxiliary feed water pump, 42 – high-pressure regeneration, 43 – cooling tower for circulation water, 44 – CV pumps).

1.2.2 Main Safety Modifications

The Equipment Refurbishment Program:

In 1998, the so-called MORAVA (**MO**dernization - **R**econstruction - **A**nalyses - **VA**lidation) equipment refurbishment program was approved, which followed the refurbishment projects implemented within the framework of the EDU completion. One of the main goals of the Morava program was to achieve the level of safety acceptable in the EU and within the framework of the project; a number of safety-relevant assignments addressing the so-called MAAE safety findings assessed within the EU as well were implemented. The implementation is associated with a significant decrease of the reactor core meltdown probability. Among the significant safety-related modifications there are:

- protection of the TQ sump - modification of the sieves as the prevention of the spray pumps´ intakes´ clogging by debris in the course of accident
- modification of the PoE equipment +14.7 m - protection of the high-energy pipelines against missiles
- complete refurbishment of the instrumentation and control (automatic systems RTS, ESFAS, ELS, RST, PAMS, etc.)
- relocation of the dividing header of the emergency feedwater pumps - separation of the safety system from the high-energy pipelines
- completion of OVKO, including the TNR protection against cold pressurization
- completion of emergency venting for TNR and the PG primary headers
- drain pipeline from A,B 301/1,2 – prevention of the loss of coolant in case of leak on the HCČ board
- modifications to improve fire protection - extension of SSZ, installation of fire-resistant doors, fire suppression on the NCČ board
- modifications of the “leak before break” design - upgrade of the KO body against vibrations, completion of accessories of the circulation pipe whip restrains
- qualification of PVKO and OVKO for work with water media
- qualification of PV PG and PSA on the main steam lines for work with water media
- implementation of nozzles onto the SHNČ pipelines enabling connection of the HZSp´s mobile pumps

- replacement of HNČ for the ones with higher capacity
- qualification of components important for nuclear safety
- increase of the rated thermal power of the reactors from the original 1,375 MWt to the value of 1,444 MWt
- adjustment of the TL11 ventilation route´s geometry in the containment to eliminate total losses of coolant during LOCA. This action comprises preparation of the suction opening for the possibility cooling TNR from outside
- implementation of the symptom based emergency operation procedures (EOPs) and guidelines to manage severe accidents (SAMG)

Periodic Safety Review:

In the Dukovany NPP, a PSR after 20 years of operation was performed in the period of 1/2006 – 6/2007. The process of the PSR EDU processing was managed in the manner allowing preserving general compliance with the IAEA NS-G-2.10 guideline and the WENRA principles for PSR defined in the PSR EDU document "Reactor Safety Reference Levels". The PSR EDU was performed to the decisive extent by employees of ČEZ, a. s., which is in compliance with the aforementioned IAEA NS-G-2.10 guideline.

Using a thorough review of the key safety-affecting areas´ condition, the purpose of the PSR according to the IAEA NS-G-2.10 guideline was

- to determine, to which extent the nuclear power plant fulfils the current internationally recognized safety standards and practice,
- to verify validity of the licensing documentation,
- to determine whether the respective actions to maintain the nuclear power plant´s safety are applied up to the next PSR,
- to determine improvements in the area of safety, which should be implemented to solve identified safety deviations.

The documentation for the purpose of the PSR´s performance was prepared in advance containing in particular the Methodologies and Criteria based on the applicable legislative documents of the Czech Republic and the applicable IAEA documents up to the Safety Guide level, documents of the INSAG line and the WENRA requirements for PSR. The review was performed for all areas (14 areas) and for all safety factors defined according to the IAEA NS-G-2.10 guideline.

The discovered deviations were categorized from the point of view of the safety relevance and using the approved Methodology, they were divided into 4 groups (high, medium, low, very low).

Actually, the comprehensive review within the PSR's framework has identified similar opportunities to increase safety as those stated in this report.

Some of them have been implemented already or are nearly completed (qualification of the EDU project, including seismic upgrade, completion of the PAMS system, creation of the seismic PSA, actions taken against irreversible loss of coolant, measurement of the level in the reactor cavity, completion of accessories of the circulation pipe whip restrains, improvement of accuracy of the containment´s integrity, processing of documentation for TPS, monitoring and evaluation of the human factor´s quality and safety culture), others are in the stage of preparation for implementation and would be implemented regardless of this new review. In the area of ageing management, this concerns implementation of the program of the service life and equipment reliability management; in the technical area, upgrade of the EDU´s design to cover consequences of severe accidents (comprising increase of the hydrogen disposal system´s capacity, completion or recombiners for severe accidents and preparation for the reactor cavity´s flooding, installation of the internal seismic monitoring system, the BD´s HVAC system´s modification and upgrade or limitation of load applied on the 12th floor of VBK, EPS´s refurbishment, upgrade of the II.O pipes in the containment, upgrade of the shelters - the Dieselgenerators and the oxygen recovery system) is concerned. In the administration and human resources area, completion of preparation of the so-called Shutdown SAMG and preparation of the means enabling graphic representation of the behaviour of severe accidents as a tool for preparation, education and training of the staff, completion of safety analyses of beyond design basis accidents and zero-power conditions are concerned. In fact, PSR assumes the implementation of the approved actions within 2015, i.e. within the time limit for the Dukovany NPP´s service life´s extension.

1.3 Systems for providing or supporting main safety function

Familiarization with the content of this chapter is necessary for correct understanding of the following chapters dealing with review of resistance for individual reviews

Design Safety Concept

Safety of the Dukovany JE 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

- Defence in depth
- Fulfilment of safety functions

Defence in depth

Defence in depth has two principal assignments:

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

The fulfilment of defence in depth within the JE project is realized by means of the following provisions:

- Five levels of defence 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 defence 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 defence in depth, the following level thereof is activated.

Fulfilment of safety functions

Functions, namely both the operational and the protection and safety ones, are fulfilled by the technological systems, structures and components.

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

- Important from the point of view of nuclear safety (participating in the fulfilment of safety functions).
- Unimportant from the point of view of nuclear safety (do not fulfil 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).

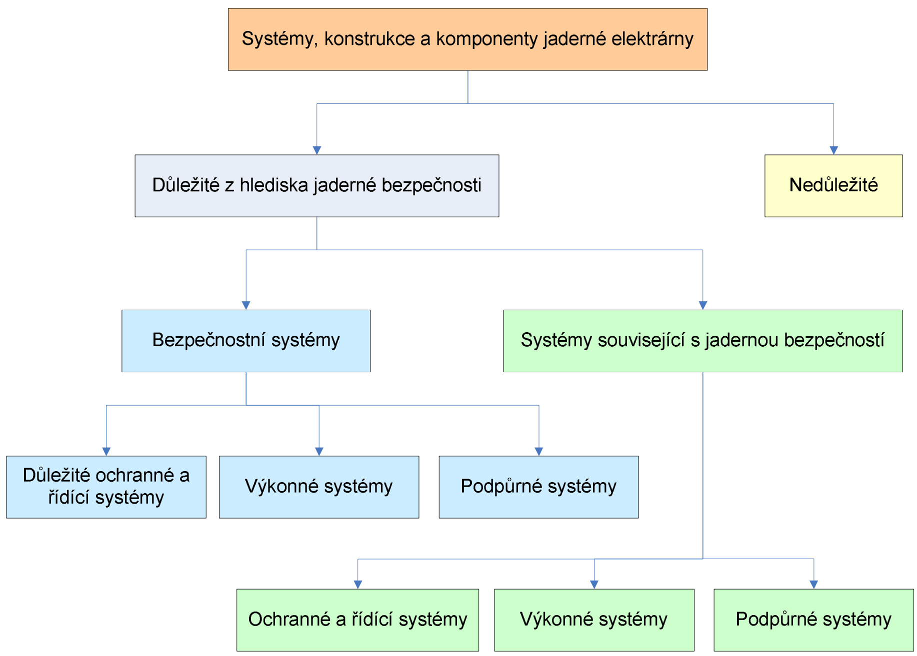
- Performance safety systems (systems which perform the respective safety functions by initiation of the safety systems).
- Auxiliary systems (systems providing for the functions of the protective and performance systems, such as power supply, cooling systems, etc.).

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 fulfil any safety function, however, may be used in case of accident conditions (if they are available).

A diagram of the SCC classification into individual categories is provided in the following figure:



(Legend: by lines:
Systems, structures and components of the nuclear power plant,
Important from the point of view of nuclear safety Unimportant
Safety systems Safety-related systems
Important protective and control systems Performance systems Auxiliary systems
Protective and control systems Performance systems Auxiliary systems)

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.

Units of this type can be characterized by the ability to provide for the basic safety functions by means of the following multiple diversified systems under the modes of normal and abnormal operation and under accident conditions:

- Sub-criticality is provided by means of mechanical regulation - by a fall of the regulation fuel assemblies due to own gravity, by means of the high-pressure active safety systems with high concentration of boric acid, the low-pressure active and passive safety systems with the shut-down concentration of boric acid and by means of the safety-related volume and boron control systems under normal and abnormal operating conditions.
- Heat removal is provided under normal as well as abnormal operating conditions by the steam generators with a large reserve of water; heat removal from the turbine condensers is provided by the circulation cooling water circuit with natural-draft cooling towers. Heat removal is further provided by the turbine bypass steam dump, reduce station, technological condensers - all of them are included in the safety-related systems. As the replacement of the aforementioned, the following safety systems are further used - the steam dump or the safety valves of the steam generators, the low-pressure active emergency cooling system with heat removal by means of TVD with the 3 x 100 % redundancy with the 100 + 100 % redundancy of active elements (pumps) inside each of the equal divisions. Generally, the safety systems are solved for each individual unit, the essential service water is solved based on the twin-unit design in this type of power station. Heat removal from ESW to the atmosphere is realized by means of independent overflow of water in the cooling towers shared with the circulation cooling water system.
- Elimination of radioactive media releases from the core is provided by physical barriers - the fuel matrix and cladding, pressure limit of the primary circuit, containment with maintenance of underpressure inside under normal and abnormal conditions. Heat removal from the spent fuel pool under normal and abnormal conditions is provided by means of the redundant (2 x 100 %) system related to safety; under accident conditions, with the use of the boric acid reserve stored in the low-pressure safety system tanks with evaporation to the area of the reactor hall. Under accident conditions, isolation of the hermetic zone from the surrounding environment is further activated together with closure of the quick-acting valves on its border with the heat removal system and decrease of pressure in the containment using the active safety spray system and heat removal using TVD to the cooling towers.

More detailed descriptions, including the method of beyond design basis solutions are provided in the following chapters.

<u>Compliance of the plant with the current licensing basis</u>

The legislative requirements imposed on a design of a nuclear facility to provide for nuclear safety, radiation protection and emergency preparedness are included in Decree of SÚJB No. 195/1999 Coll. The EDU design complies with the requirements imposed by the aforementioned Decree and it has been demonstrated by means of analyses that during all expected states of normal operation, abnormal operation even under emergency conditions, the ultimate parameters of fuel are not exceeded and that during design basis accidents, no exposure of the critical group of population, which would approach the rejection criteria defined by Decree No. 215/1997 Coll. and for which early implementation and full realization of all urgent actions to be taken to protect members of the public would be unavailable, will occur.

The deterministic review of the Dukovany NPP´s operational safety (licence analyses) is based on the IAEA-EPB-WWER-01 guideline *Guidelines for Accident Analysis of WWER Nuclear Power Plants, Dec. 1995*, which specifies the scope of ***initiating events***, for which the fulfilment of the ***acceptance criteria*** of the NPP´s operation and the requirements imposed on the ***methodology*** of such analyses´ performance must be proven.

According to probability of occurrence (frequency), initiating events are categorized as follows:

1/ ***Expected transients*** – the probability (frequency) of occurrence thereof is within the range of $1 \div 10^{-2}$ reactor-year. For the aforementioned ones, stricter acceptance criteria (= more distant from the actual value of the parameter in view when safety is becoming impaired) and the fulfilment thereof must be demonstrated even in consideration of the pre-defined conservative (= tightening) prerequisites.

2/ ***Postulated incidents*** *(initiating events)* – the probability (frequency) of occurrence thereof is less than 10^{-2} reactor-year. For such events, even less strict acceptance criteria (= less distant from the actual value of the parameter in view when safety is becoming impaired) may be defined and fulfilment thereof is required:

- o within the range of probability /frequency) of occurrence $10^{-2} \div 10^{-4}$ reactor-year upon fulfilment of the conservative prerequisites similarly as in case of transients
- o for events having the probability (frequency) of occurrence lower than 10^{-4} reactor-year, application of even less strict acceptance criteria is allowed and application of lighter (= less strict) conservative prerequisites or application of realistic (= without conservative prerequisites), the so-called *Best Estimate* analyses.

As an example of determination the acceptance criteria, the construction of the acceptance criterion for radiological consequences of the respective initiating events (PpBZ EDU rev. 2, chap. 15.0)
The initiating events are further divided into groups/sub-groups, according to the type (character) of the event, and performance of analyses of at least one initiating event from every group/sub-group is required.

Inspection of the acceptance criterion: “fuel temperature shall not exceed the value of 2,520 °C in any location of the core” secures that in the course of any of the analysed initiating events no local fuel melting would occur.

1.3.1 Reactivity control

1.3.1.1 Sub-criticality of the core

The reactor core consists of 349 assemblies out of which 312 are operating and 37 are HRK (accident and regulation fuel assemblies) that can be re-located vertically.

The arrangement of the operating fuel assemblies and HRK in the core is in a triangle storage rack with distance of 147 ± 0.6 mm.

The step of the storage rack provides self-regulating characteristics of the core when working at the output due to negative feed back of the output and moderate temperature and fuel (coefficients of reactivity from output and temperature are negative)

It is stable against radial and axial fluctuation due to the relatively small core when the output is distributed at transients from xenon.

There are two independent systems based on different technical principles to control the reactivity:

- Mechanical system to trip reactor including supply breakers
- System of adding and boron regulation (CVCS)

The mechanical system of reactor trip belongs to the safety system (BS). It consists of accident and regulation fuel assemblies and its functions are as follows:

- it ensures quick breaking of the chain reaction in the reactor by quick fall of the absorption part into the core and at the same time ejecting the fuel part out of the core

- it is involved in the automatic regulation with the aim to keep the output of the reactor at the level set and in transition from a output level to another

- it compensates quick changes in reactivity.

In total, there are 37 HRK divided into six groups in the reactor.

The safety function of the quick trip of the reactor is fulfilled by fall of the HRK due to own gravity after the supply breakers are switched off. Loss of power supply means automatic switch-off of the breakers (i.e. safe accident).

System of adding and boron regulation (CVCS)

a) System of adding belongs to the safety-related systems (SSB). There are two basic operation systems TK, TE.

TK system of common adding and boron regulation intended for:

- water supply to rod wiper of the HNČ in all operation modes of the power plant

- adding the primary circuit (compensation of non-organized leaks)

- returning the organized leaks to the primary circuit in all operation modes of the power plant

- to compensate the changes in reactivity of the reactor by supplying boron concentrate or pure condensate to the primary circuit

- to maintain the level in the KO when changing the output of the reactor and to cool the KO

- to correct chemical regime by supplying chem. reagents to the primary circuit

- filling the primary circuit

- the system can serve also for primary circuit make-up in accident situations (leakage of the primary circuit, breakage of the steam pipeline, breakage of PG tube) even though it is not primarily intended for these accident situations

It is a technological circuit with pumping sets of 3 x 100% power supplied from category II of the secured supply (DG).

Main equipment of the system:

- deaerators of adding TK 10, 50B01

- adding sets, consisting of pumps TK20, 40, 60D01, 02 – 3 pcs

- 2 independent routes

TE system of the primary circuit coolant discharging intended for:

- discharging the coolant from the primary circuit in all operation modes

- coolant removal to be deaerated

- deaerated coolant removal to the cleaning station of drain water of the primary circuit ev. to the tanks of impure condensate

- solution removal after de-activation of the internal surface equipment of the I.O

It is a technological circuit with an operating and back up pump power supplied from category II of the secured supply (DG).

Main equipment of the system:

- discharge pumps TE(10,50)D01
- 2 independent routes

b) The system of boron regulation belongs to the SSB. There are two basic operation systems TC, TB.

TC system – A system of continuous primary circuit water cleaning belongs to the SSB.

The system is mainly intended for maintaining the required quality of the primary circuit coolant defined by the standards on water-chemical regime.

The system of the continuous primary circuit water cleaning consists of two in terms of function equal and fungible circuits TC10 (TC11N01 – katex, TC11N02 – anex) and TC50 (TC51N01 of mixed type); one circuit is operating, the other one is back up.

TB system – A system of adding boric acid to the primary circuit belongs to the SSB.

The aim of the system is to store the H_3BO_3 concentrate and adding it to the primary circuit. The system (junction) consists of two storage tanks of H_3BO_3 concentrate (2x50m^3) and six pumps (1 low-pressure (handling), 2 high-pressure (used for pressure tightness test), 3 emergency – supplied from II category of the secured supply (DG). The H_3BO_3 concentrate is distributed to every of the two storage tanks through separate routes from the concentrate cleaning station and from the chemical reagent plant.

Apart from the systems described above, the emergency systems of the cooling of the core, belonging to the BS, are intended for providing safety functions to reactivity control, too.

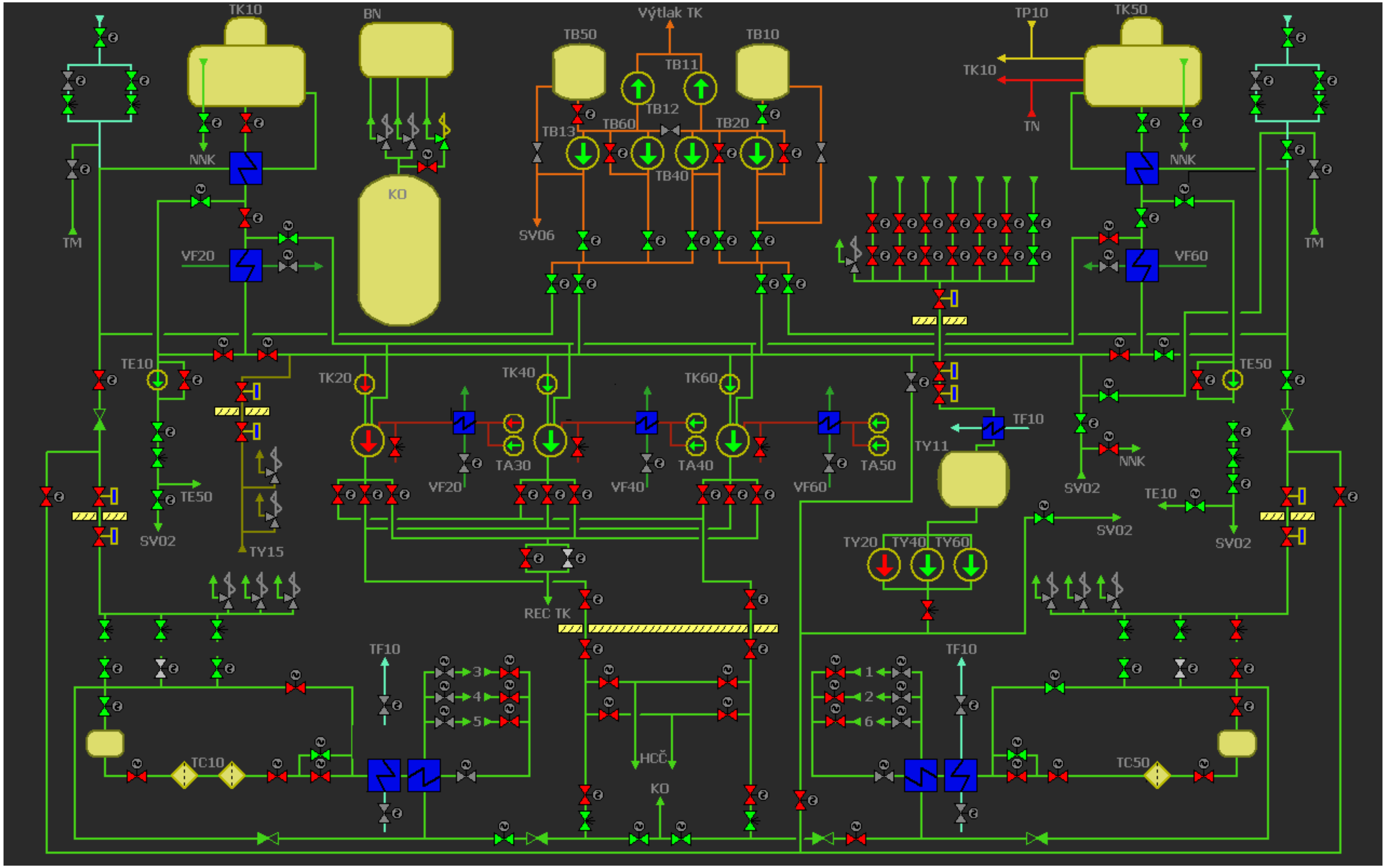


Fig.1.3.1.1-1 System of adding and boron regulation

Emergency cooling systems of the core (high-pressure TJ, low-pressure TH) - active

a) TJ system – A high-pressure system of emergency cooling of the core is used to mitigate the course of and liquidate the consequences of accidents related to loss of tightness of the primary circuit eventually the secondary circuit. During the common operation of the unit at the nominal or lower output, the equipment is in stand-by mode being ready to automatically start if an accident occurs.

In case of an accident, it provides:

- adding to the I.O and increase in the concentration of the boric acid in the I.O if the I.O is not tight or if the II.O is broken in order to limit the damage to the fuel

- prevention of not admissible transients related to the changes in reactivity

-, through its function and with other safety systems, limitation of leaks of radioactive substances and of release of radiation dose from the hermetic zone in emergency conditions and afterwards.

The system is solved using the system redundancy of 3 x 100% including all supportive systems (cooling, power supply, control and ventilation).

The TJ pumps are power supplied from category II of ZN.

The main equipment of the TJ system of the emergency cooling:

- high-pressure emergency adding pump – 3 pcs – flow of 65 m^3/h at counter-pressure of 12.7 MPa, max. pressure at pressure discharge of 14.3 MPa

- tanks of the TJ system - 3 pcs (3x80 m^3 H_3BO_3 of concentration of 40 g/kg and of temperature of 50℃)

b) TH system – a low-pressure system of emergency cooling of the core is used to mitigate the course of and liquidate the consequences of accidents related to high leaks from I.O. During the common operation of the unit at the nominal or lower output, the equipment is in stand-by mode being ready to automatically start if an accident occurs. The TH system is designed in a similar way like the TJ system (i.e. with redundancy of 3 x 100%).

The TH pumps are power supplied from category II of ZN.

The main equipment of the TH system of the emergency cooling:

- low-pressure emergency adding pump – 3 pcs – 280 m^3/ h at pressure of 0.71 MPa at pressure discharge

- tanks of the TH system – 3 pcs (3x250 m^3 H_3BO_3 of concentration of 12g/kg)

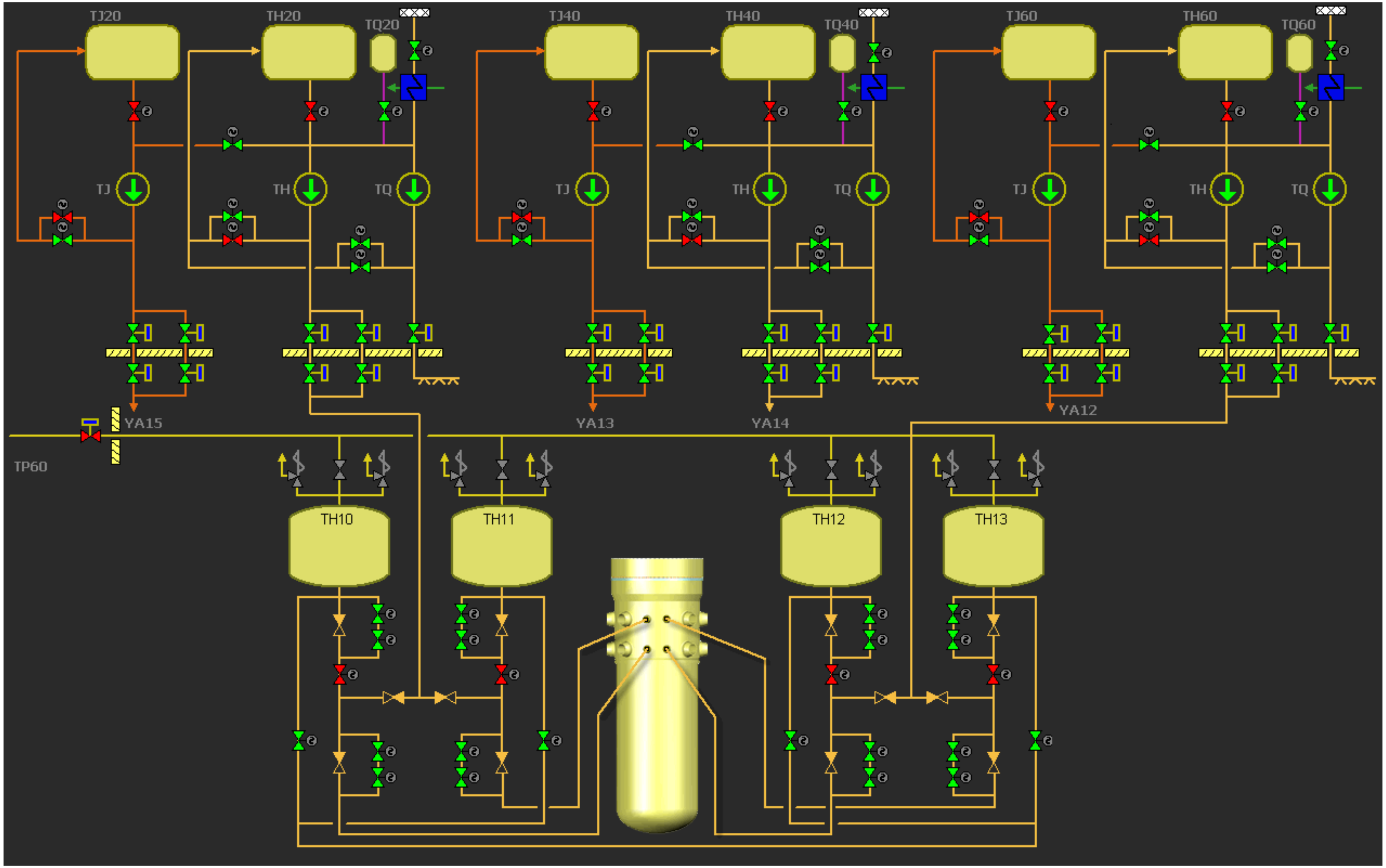


Fig.1.3.1.1-2 Emergency cooling systems of the core

Emergency systems of adding TH - passive

The system of pressure tanks is intended for sealing the core in the initial stage of accidents and it provides quick sealing of the core with the solution of boric acid. The system consists of 4 pressure tanks (4x40 m^3 H_3BO_3 of concentration of 12g/kg). Each of the four pressure tanks has its independent discharge route DN 250 directly to the reactor. The storage tanks are split into the reactor thanks to the nitrogen cushion in a passive way by decreasing the pressure in the I.O.

In order to ensure reactivity control in case of beyond design basis accidents, apart from the systems described above, it is possible to use other systems - TM, TD - too.

TM system – a system of cleaning the pool water belonging to the SSB.

The system is intended for clearing the water in the tanks of the emergency reserve of the solution of boric acid of the bubbler condensate, emergency cooling system of the core and spent fuel storage pools from the soluble and dissoluble impurities. For the purposes of providing the safety function of the reactivity control, it is possible to use the unit pumps of this system TM13, 14D01 power supplied from the secured supply of category II (DG) in order to pump the solution of boric acid between the NT tanks of the emergency system and then it is supplied to the primary circuit after it is de-pressurized.

Analogically, it is possible to use the unit pumps of the system TD60 to pump the concentrated solution of boric acid between the VT tanks of the emergency system and then it is supplied by means of the high-pressure pumps of the emergency systems to the primary circuit system.

1.3.1.2 Sub-criticality of the spent fuel storage pool

Sub-criticality of the fuel assembly set in the storage pool is provided by two independent ways:

- geometry and material implementation of the storage racks located in the spent fuel pool
- concentration of boric acid in the pool

699 spent fuel assembly sets (PS) out of which 17 for PS located in hermetic cases (HP) can be stored in the basic compact storage rack. When there is outage with full fuel removal from the reactor, the so-called back up storage rack is temporarily located in the BSVP above the basic storage rack i.e. at level of +10.97 m. The back up storage rack enables storage of 296 pcs of PS and 54 pcs of HP.

When storing the fuel in two layers, the minimum level of +18.5 m is prescribed, when handling the fuel the minimum is +20.75 m.

The concentration of boric acid in water is maintained at the minimum value of 12 g/l. The water temperature of the system of ≤ 60 ℃ is prescribed for BSVP in accordance with the Limit and Conditions (LaP); the temperature is to be maintained under 50℃ in accordance with the operation guidelines (PP).

There are no restrictions for placing the fuel in the bottom basic compact storage rack. For the purposes of placing the fuel in the upper, i.e. back up, storage rack, guidelines have been prepared for placing the PS so the PS with fresh fuel are surrounded with the more spent PS.

Sub-criticality is provided in the conditions aforementioned even if the spent fuel pool is filled with pure condensate, i.e. $\mathbf{k_{ef}<0,95}$.

1.3.2 Heat removal from reactor to the ultimate heat sink

7 modes of the unit are distinguished for the nuclear unit VVER-440 type 213.

MODE	Name of MODE	Reactor output [% N_{nom}]	Reactivity $\Delta k/k$ [%]	Temperature $T_{I.O}$, T_{TNR}, T_{HVS} [°C]	Pressure $p_{I.O}$ [MPa]
1	**Operation at output**	> 2	> -1	$T_{I.O}$> 250	$p_{I.O}$ > 8.3
2	**Non-output operation** τ_R = 72 hours	≤ 2	≥ -1	$T_{I.O}$ > 190	$p_{I.O}$> 8.3
3	**Hot reserve**	Residual output	< -1	T_{HVS}≥ 180℃	$p_{I.O} > p_{atm}$
4	**Half-hot reserve**	Residual output	< -1	T_{HVS}≥ 90℃	$p_{I.O} > p_{atm}$
5	**Outage with cooling the I.O**	Residual output	< -1	T_{HVS}< 90℃	$p_{I.O} > p_{atm}$
6	**Outage with unsealing the I.O**	Residual output	< -1	T_{HVS} < 100	$p_{I.O} = p_{atm}$
7	**Removing the fuel from the core**	**The core does not contain fuel**			

T_{HVS} – temperature of the hot branches of the I.O

1.3.2.1 All existing heat removal means / chains from the reactor to the UHS in different reactor shutdown conditions.

These chapters describe the ways of heat removal from the core to the UHS in modes 3 to 6 i.e. modes when the unit has been shut down and it is necessary to provide heat removal from the I.O to the end heat sink up to the state when it is possible to unseal the I.O and remove the residual heat after it has been unsealed.

In order to ensure heat removal for the purpose of meeting all the above-mentioned reasons, it is possible to use both operating systems and safety systems.

Operating systems are:

- Condensation system and turbine bypass steam dump

Safety-related systems (SSB, safety related):

- <u>Cooling system</u>
- PG <u>emergency supply system</u> (together with PSA or PVPG safety systems)

Safety systems:

- <u>Super-emergency PG supply system and PVPG and PSA system</u>
- <u>Cooling using SAOZ</u>

Other systems

It is use of standard design means beyond their design designation e.g. in the field of beyond design basis accidents. It is possible to use the so-called non-standard solutions e.g. operating and safety systems working together at the same time to remove heat from the core. The way of residual heat removal by means of water supply to the PG using the equipment of the fire brigade (HZS) is described in the EOP guideline mainly.

1.3.2.2 Lay out information on the heat removal chains: routing of redundant and diverse heat removal piping and location of the main equipment. Physical protection of equipment from the internal and external threats.

Operating systems

<u>Condensation system</u>

The common operating system for heat removal from the core to the end heat sink is to remove heat through the secondary circuit when heat is removed by means of forced circulation (if HCČ are in operation) or natural circulation from the core. Heat is removed from the I.O in PG; the steam is removed to the TG condensers. Heat is removed by means of circulation cooling water from the condensers. Water is removed by means of condensation pumps that provide water to feed tanks from the condensers. Water is supplied by means of common or emergency PG supply from the feeding tanks to PG.

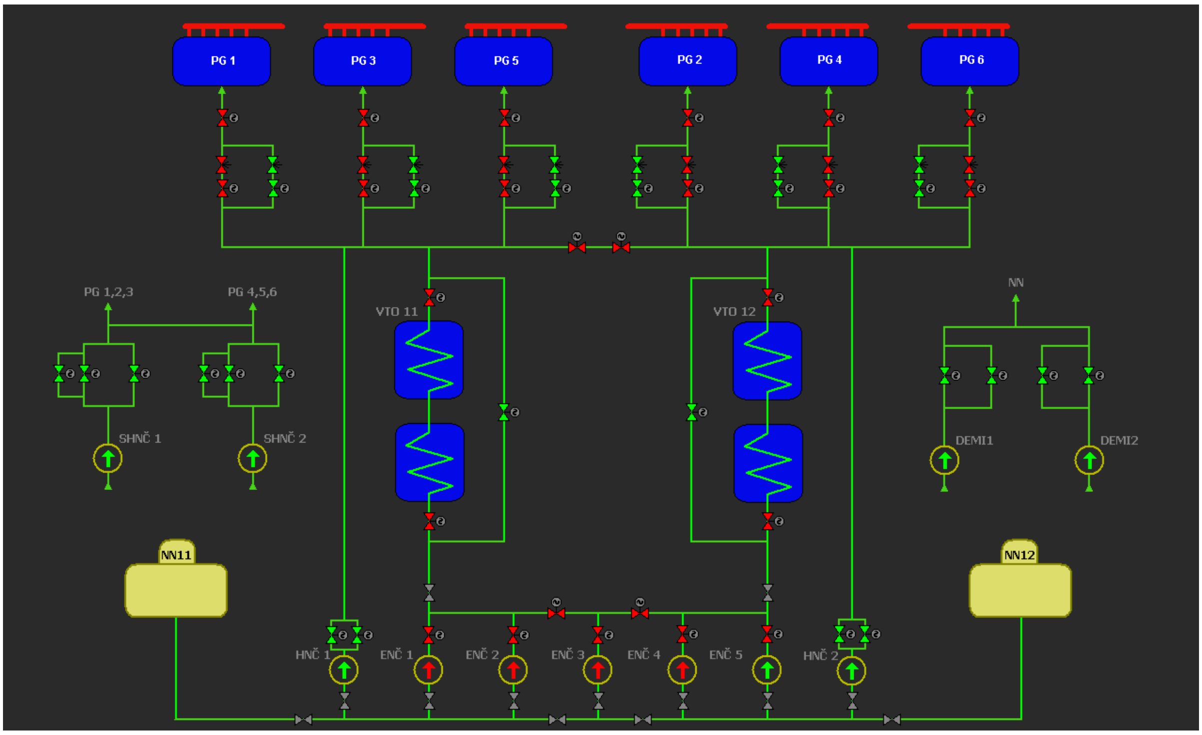
Cooling system

This system operates in two operation modes - steam and water. Heat removal in both these modes is provided by means of the ESW in the technological condenser (hereinafter “TK"). Heat removal from the core to PG is provided by means of forced or natural circulation.

In steam mode steam is removed through the reduction station of the cooling system from PG to TK, from there it is removed by means of the ESW. The condensate is removed to SNK, resp. NN. The water level in PG is provided by means of the system of common or emergency PG supply. Demi water pumps 1 MPa add eventual lost water in the system.

NN and demi water tanks are the sources of water; pumping is provided by means of ENČ or HNČ power supplied from SZN, category II (DG); heat removal from the TK is provided by means of the ESW pumps, power supplied from SZN, category II; the demi water pumps are power supplied from SZN, category II (DG).

All the systems mentioned (except for the ESW) belong to the SSB. The ESW belongs to the BS.



In water mode, TK is used as exchanger water/ water. Steam pipelines and HPK routes and routes of the whole cooling system are filled with water by means of the HNČ. The pipeline routes including the suspenders are designed for water filling. After the routes are fully filled, the coolant circulation is provided by means of cooling pumps. Eventual lost water in the system is added from the DV tanks. At least a NN fulfilling the function of volume compensator is used, too.

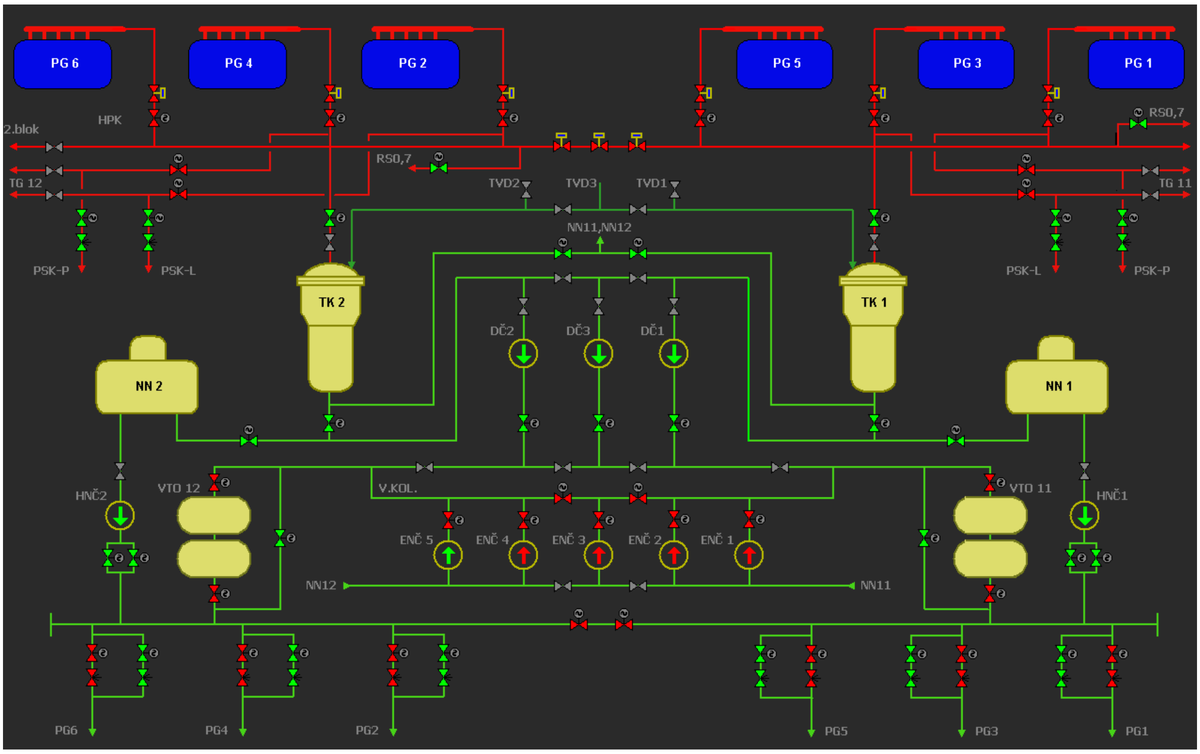
All these systems belong to the SSB; demi water tanks 1000m^3 belong to the BS; RSD, technological condensers and pipeline routes are doubled (2x100%), the pumps are 3x100%. The system is located in the turbine hall and it is not separated from the common operating systems in terms of the lay out. The supply routes to the PG including the regulation and closing valves are located in the longitudinal interim room on floor +14.7 m; the steam pipelines from the individual PG including the valves are located in the same room. Each of the technological condensers is cooled with the other division of the ESW. The third division of the ESW that can be connected to both TK is used as a back up. There is no seismic design of the cooling system; only the technological condensers are of the Sc design, i.e. resistance to tipping. Protection against external events is same like for all other systems located in the turbine hall (see other chapters of this report).

Emergency PG supply system

The emergency supply system consists of two emergency supply pumps (HNČ) and doubled pipeline routes (2x100%). They suck from the feed tanks (NN); discharge routes are led outside the high-pressure heaters (VTO) and they are led directly to the emergency supply collector (HNK). Separate routes of emergency supply of smaller diameter with closing and regulation valves have been created for the PG supply. These routes are interconnected with the routes of the common PG supply and the containment and PG supply is led through the common pipeline for every PG. The system is supplied from ZN category II and the pumps are started with the ELS programme. The system is located in the turbine hall and it is not separated from the common operating systems in terms of the lay out; the routes are connected to the common PG supply system. The system is used for residual heat removal by means of the technological condenser in steam mode, for PG supply at low power (when TG are out of service) and it is also possible to use it for PG supply in secondary feed&bleed mode removing steam through PSA or PVPG.

The ESW is not needed in the secondary feed&bleed mode; it is thus possible to remove residual heat on a long-term basis in this way even if there is loss of UHS. The pumps and valves are power supplied from ZNII; PSA and PVPG are power supplied from ZNI so it is possible to remove heat in this way even if there is ÚZNVS, i.e. when DG are operated. However, if DG are operated, the ESW system is essential for DG cooling. The heat is removed from the reactor by means of natural circulation in ÚZNVS mode and the operators shall follow the EOP guidelines, ES-0.2. Procedure. If steam is removed through PSA, it is possible to cool the temperature in the I.O to c. 120-130 ℃ in 30-40 hours after the reactor has been tripped. In order to remove residual heat (without cooling), the flow of feed water to PG of 37 m^3/h is needed to be reached in 0.5 h after the reactor has been tripped, c. 10 m^3/h in 40 h, if cooled by 10℃/ h in natural circulatio n mode, c. 17 m^3/H in 40 hours. The feed water is supplied to PG from the NN; the NN is added from the demi water tanks (3x1000 m^3 per HVB) by means of demi water pumps 1MPa. If the ES-0.2 is followed, the reserve demi water in the tanks of 1000 m^3 considering the current residual heat removal from both reactors within a HVB shall be sufficient for 3 days, then there is water in the NN available. Demi water tanks can be added from the available water reserve from the EDU premises (clarifiers, cooling tower pools) by means of mobile technology; stable junction pieces have been installed for this purpose so it is easy to connect the fire hoses.

The secondary feed&bleed mode can be used even in water cooling mode i.e. after PG and steam pipelines are filled with water; water would be discharged through PSA or PVPG in this mode. Analyses have been carried out for this way of cooling, however, operation guidelines have not been prepared yet. Higher flow of water to PG is necessary and it would be necessary to use both HNČ or both SHNČ for the purpose of this procedure.



Safety systems

If the systems mentioned above are not available, there are systems belonging to BS available.

Super-emergency PG supply system and PVPG and PSA system

Heat is removed by means of forced or natural circulation from I.O. Heat removal in the II.O is then carried out through the not closed circuit i.e. through the turbine bypass dump to the atmosphere resp. through PV PG to the atmosphere.

Water supply to PG if the ENČ or the emergency pumps HNČ are not available can be provided by means of the SHNČ pumps.

Three demi water tanks standing in the premises at the wall of HVB are the water source for the SHNČ; 2 units within a HVB share the tanks. The water temperature in the tanks is c. 20℃, protected against freezing in winter. There a re two SHNČ pumps for every unit; they are located in a separate building inside the HVB with the demi water pumps 1 MPa. The SHNČ pumps are located in separate boxes and thus they are separated from each other in terms of the lay out. The SHNČ pumps and their routes are doubled (redundancy 2x100%). The pumps and valves are power supplied from ZN II; the pumps are started by means of the ELS programme and ESFAS signals, too. The discharge routes are led through the pipeline channels to the turbine hall and from there to floor +22 m to the longitudinal interim room. The feeding routes with closing and regulation valves are separated from each other in terms of the lay out; the routes for PG1, 2,3 are located in a room and the routes for PG 4, 5, 6 are located in another separate room. From there the routes go to floor +14.7 m and they are led through the hermetic penetrations to the PG box and to individual PG. The pipeline routes of the super-emergency supply are protected with massive armour plating against flying objects in the turbine hall and longitudinal interim room. The SHNČ routes are entirely separated from the routes of the common PG supply in terms of the lay out and they are led to PG through separate junction pieces.

The equipment of the SHNČ system is of the seismic design with the minimum resistance of SL2; the building of the SHNČ and demi water tanks is seismic-resistant and also resistant to extreme weather conditions.

PSA and PV PG are located in the longitudinal interim room of the HVB on floor +14.7 m. PSA are located at the HPK in 2x100%; there is a pair of PVPG on every PG. PSA and PVPG are power supplied from ZNI; remote control is possible from BD and ND; PSA have a manual wheel to be opened on the spot. PSA and PVPG are seismic qualified and HELB-condition qualified; the medium removal is possible both in steam and water mode. PSA enable to lower the pressure virtually to the level of the atmospheric pressure in the II.O; PG can be de-pressurized by means of PVPG to pressure of 3.5 MPa only. Currently, a modification to lower the pressure by means of PVPG to the atmospheric pressure is prepared.

When using the SHNČ system, demi water of temperature of c. 20℃ is su pplied directly to PG. A detailed procedure ES-0.6 has been prepared within EOP for the purpose of using the SHNČ. In order to remove residual heat, the ESW is not needed thus it is possible to use the procedure of the secondary feed&bleed in case of UHS loss, too. If, however, UHS loss is related to ÚZNVS, we need the ESW to cool the DG. When cooling by less than 10℃ in the mode of natural circulation, demi water reserve in tanks 3x1000 m^3 will be sufficient for a period longer than 72 hours at the current residual heat removal from both reactors of a HVB. Demi water tanks can be added from the available water reserve from the EDU premises (clarifiers, cooling tower pools) by means of mobile technology; stable junction pieces have been installed for this purpose so it is easy to connect the fire hoses.

Cooling using SAOZ

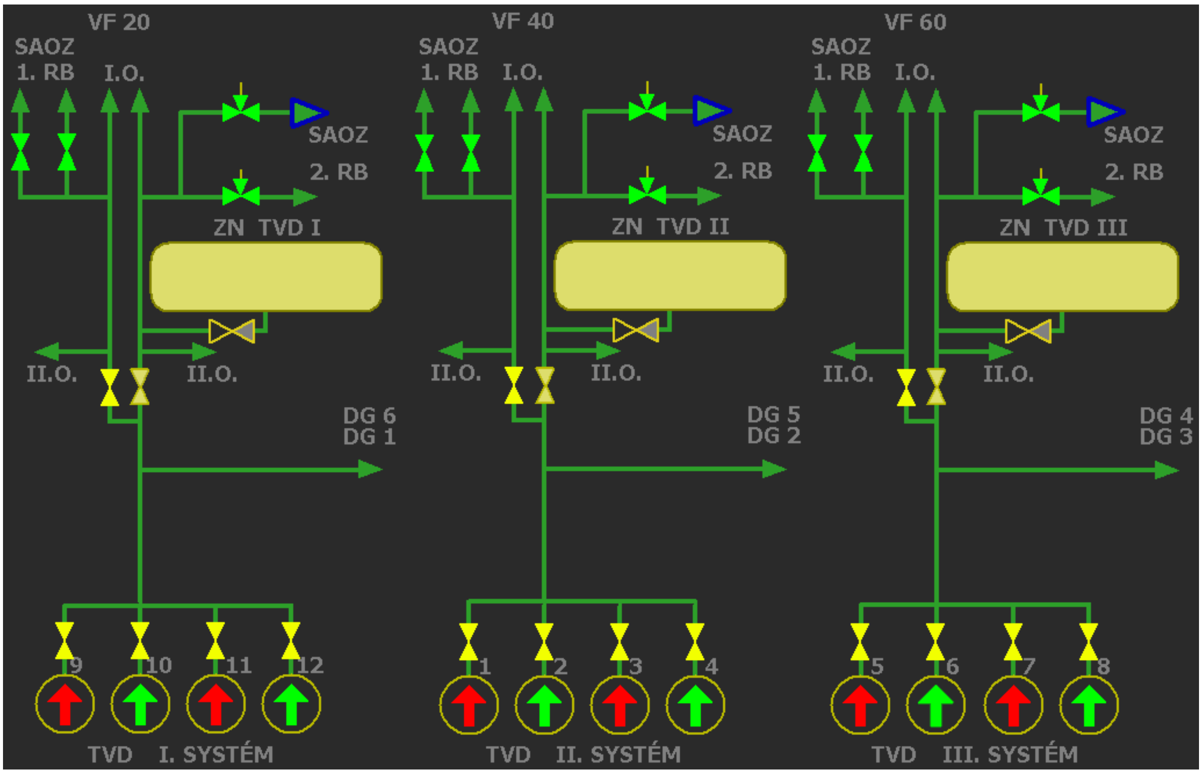
Other possibility to remove heat from the I.O (except for the forced and natural circulation considered as described above) is the usage of the VT emergency system by means of the TJ pumps discharging the coolant from the I.O to the containment through the OVKO, resp. PVKO in Feed&Bleed mode. Heat removal would be provided from the I.O to the containment in this mode and from there through the SAOZ exchanger by means of the ESW system. Moreover, the TQ spray system would be used to maintain the pressure in the containment and steam condensation. The VT pumps are able to supply concentrated boric acid solution to the I.O at the nominal pressure. The mode has been analysed at the operation of one, two TJ pumps and opened OVKO or PVKO and their combination. If the cooling of the core fails to be restored by the II.O and if low pressure is reached in the I.O., it is possible to use the low-pressure pumps of the TJ system instead of the TH pumps.

All BS pumps are power supplied from category II of SZN. The SAOZ equipment is located in the system rooms of the HVB separated from each other

ESW systems

The ESW system is of key importance from the point of view of provision of safety and residual heat removal either from the fuel located in the core or from the fuel deposited in BSVP to the ultimate heat sink.

With respect to the 3x100 % ESW systems redundancy and the 2x100% additional internal redundancy of each ESW division (4 pumps), the loss of ability to remove heat from the sources is conditioned by unavailability of all the ESW pumps (in total 12 pumps). 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 TVD pumps is highly unlikely. Even in case of operation of only one pump in one division of the TVD system, it is possible to cover the basic safety functions.



Other systems

Because it is necessary to analyse all possibilities of beyond design basis accidents as for heat removal from the core, the so-called alternative solutions being beyond the scope of the common design solutions can be used. It is a possibility of gravity filling of PG directly from the NN without using adding pumps. This mode would be used combined with steam removal from the PG through the PSA. The advantage of the mode is that it can be used at SBO, too. It is sufficient to use the PSA for the heat removal (they are power supplied from category I of the power supply and they can be opened on the spot, too). In order to use this method, it is necessary to de-pressurize the PG to the pressure lower than in the NN (0.7 MPa).

Further, it is possible to use the cooling mode using HNČ resp. SHNČ and water removal through the PSA. These systems enable the so-called Feed&Bleed on the secondary part. This method enables to cool only slowly up to temperature of c. 90℃, eventually it can be used to stabilize the temperature. It is limited by the relatively small flow of the pumps used and limited capacity of the tanks 3x1000m^3.

Alternatively, the ESW of various systems can be connected to the technological condenser.

1.3.2.3 Possible time constraints for availability of different heat removal chains, and possibilities to extend the respective times by external measures (e.g., running out of a water storage and possibilities to refill this storage).

Operating systems

Use of the cooling system cooled by the ESW system is supposed. Heat removal by means of the condensation system and circulation cooling water does not have to be always available. In case of the conservative presumption of quick trip of the reactor, the heat removal from the I.O will be provided by means of steam removal from the PG to the atmosphere through the PSA, event. PVPG and PG make-up by means of the HNČ from the NN (Feed&Bleed in the II.O). The flow of 30 t/h is sufficient to remove residual heat. Water loss in the NN is compensated by means of the demi water system 1 MPa. Heating of the cooling system from the cold state takes approximately 4 h, which means loss of water of c. 120 t in the demi water tanks. It is a negligible amount with respect to the volume of 1000 m^3.

If changed to steam cooling by means of the cooling system, the circuit closes and there are no losses of demi water in fact. It is necessary to fill the PG (6 x 24 = 144) and the steam pipeline routes (160 t) for the possibility to change to water-cooling. Adding from the demi water tanks 1000 m3 shall again compensate the loss of water in the NN. There will be just changes in the coolant volume related to the decreasing temperature in this closed circuit when the water circuit is started.

The water reserve in the NN and the reserve in the demi water tank 1000 m^3 is fully sufficient not only for cooling the unit but also for continuous residual heat removal during the water stage.

Safety systems

Super-emergency PG supply system and PVPG and PSA system

If the unit is cooled in Feed&Bleed mode in the II.O (PG power supplied by means of the SHNČ pump and steam transferred through the PSA, event. PV PG) the usage time of this mode is given by the amount of water in the demi water tanks 1000 m^3. At the current cooling of all units, the amount of water in the demi water tanks and both NN is sufficient for 4 days at least. In case of lower modes (2, 3, 4) longer spare time can be considered. We do not consider this configuration for modes 5 and 6 as for the maximum usage time is needed for operation in steam mode.

The demi water tanks 1000 m^3 can be added from the fire brigade vehicles or through hoses from the tower pools, UCHV or retention tanks. In this way, it is possible to prolong the usage time of this configuration with the SHNČ for an indefinite period of time. If the SHNČ does not work, it is possible to power supply the PG by means of the HZS pumps, too. Junction pieces for connecting the fire brigade equipment are led from the wall of the SHNČ building.

Cooling using SAOZ

If there is loss of heat removal in the II.O e.g. due to loss of the PG power supply, it is possible to remove heat from the core using the Feed&Bleed mode in the I.O. It is use of the VT emergency system with the TJ pumps in the Feed&Bleed mode discharging the coolant from the I.O to the containment through the OVKO, resp. PVKO. If there is a change to the recirculation phase, it is possible to remove heat for a long time because it is a closed cooling circuit. Heat is removed through the TQ cooler to the ESW system. In order to reach the low pressure in the I.O, it is possible to use the low-pressure pumps of the TH system instead of the TJ pumps. From the physical point of view, the Feed&Bleed mode can be used in any mode (3, 4, 5, 6).

Other systems

Emergency PG supply system and PVPG and PSA system

The Feed&Bleed mode on the II.O part can be also operated if the HNČ pumps are used whereas the steam discharge from the PG goes through PSA and PV PG like when using the SHNČ pumps. The difference is that the HNČ pumps suck from the NN tanks (the SHNČ pumps suck directly from the demi water tanks 1000 m^3.) The decrease in the level in the NN is compensated by means of the demi water pumps 1 MPa sucking from the demi water tanks 1 MPa. The disadvantage of this mode is that water temperature is c. 164 ℃ in the NN compared with the demi water tanks 1000 m^3 where the water temperature is c. 20 ℃ - 30 ℃. There is c. 2 x 150 t of coolant at the nominal level in both NN. Though it is possible to provide heat removal from the core in this mode for a period of dozens of hours. In order to determine more precise time of discharging the respective tanks, it would be necessary to carry out complementary analyses.

The demi water tanks 1000 m^3 can be added from the fire brigade vehicles or through hoses from the tower pools, UCHV or retention tanks like in the mode using the SHNČ pumps. In this way, it is possible to prolong the usage time for an indefinite period of time.

PSA combined with gravity filling of PG directly from NN

In case of a SBO event it is possible to use the gravity filling of the PG directly from the NN without using any feed pump as a source of NV. It is thus a passive way of PG power supplying. In this mode, heat removal can be provided for a period of approx. twenty hours after the SBO has occurred. This mode can be also used in case of a seismic event because the respective equipment has been considered for a design earthquake.

1.3.2.4 AC power sources and batteries that could provide the necessary power to each chain (e.g., for driving of pumps and valves, for controlling the operating systems).

The components described in the previous two chapters are power supplied from the sources of category I and II. The pumps are power supplied from category II of SZN; the PSA and quick-acting valves are power supplied from category I. The systems of check and control are power supplied from category I, too. More details on power supply in separate chapters 1.3.5 and 1.3.6.

1.3.2.5 Need and method of cooling equipment that belong to a certain heat removal chain; special emphasis should be given to verifying true diversity of alternative heat removal chains

The operating system of cooling removes heat by means of the ESW in the end. The steam and water mode can be considered two physically different ways of cooling because they use different pumps for their operation. Two technological condensers are available, each of them is cooled by means of a different ESW division with that if any of the ESW divisions is eventually lost it is possible to re-connect the third (back up) division to any of the two TK.

The safety systems on the II.O provide heat removal directly to the atmosphere (PSA configuration plus cooling with the SHNČ) in steam mode. If this equipment is power supplied from the DG in operation mode, the ESW system is needed (for cooling the DG). However, such heat removal by means of the secondary part is the simplest and executed by means of the minimum means.

VT and NT SAOZ always need the ESW for heat removal. Like in the case above, if operated from the DG, the ESW is necessary for its cooling.

Conclusions on the necessity of the ESW are applicable to the configuration of discharge of steam through the PSA and adding the feed water by means of the HNČ.

The gravity filling of the PG is a passive way of heat removal to the atmosphere; its functionality is limited by the water reserve in the NN.

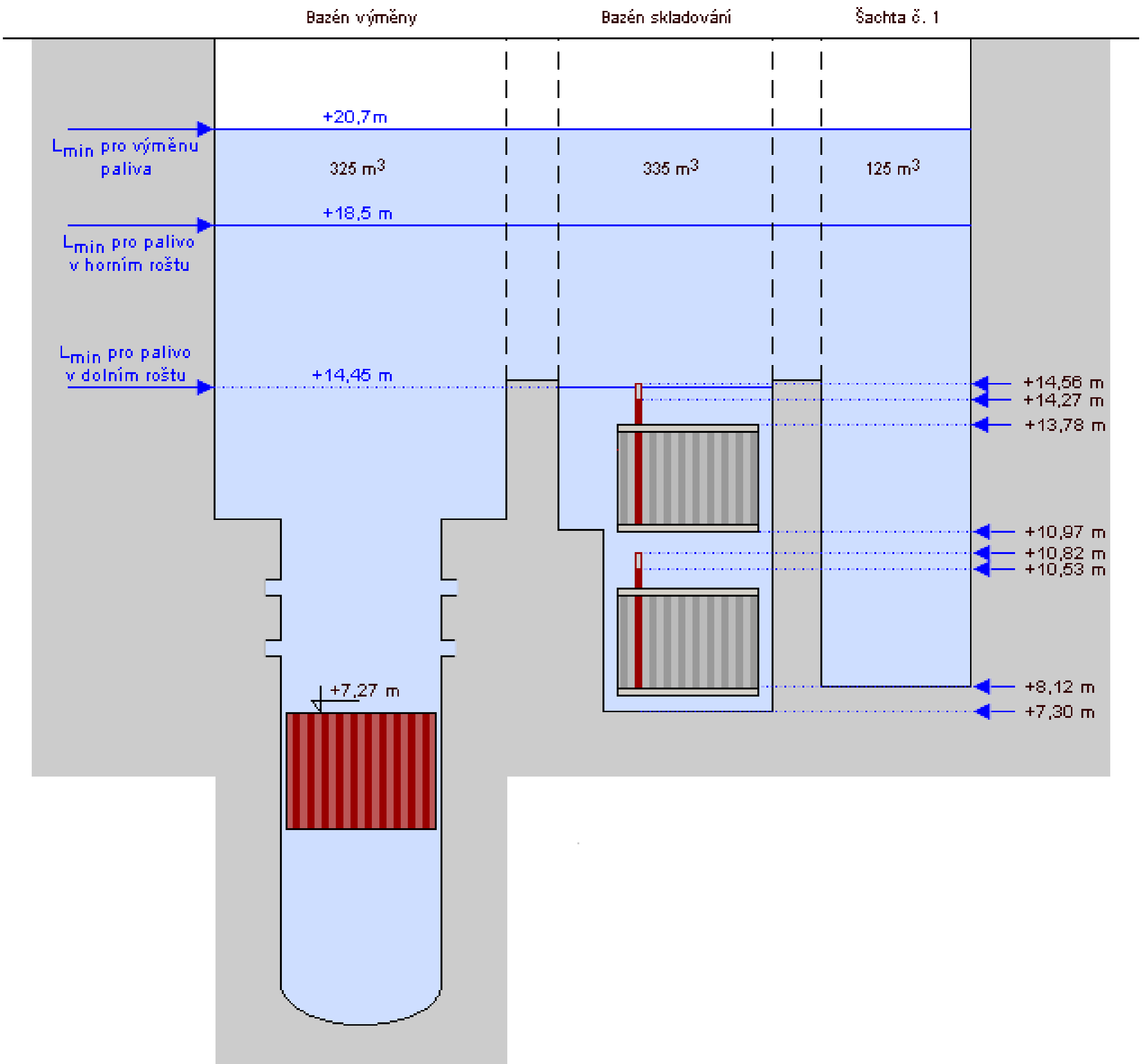
The last considered option is PG make-up by means of mobile technology (road tankers).

1.3.3 Heat removal from spent fuel pool to the ultimate heat sink

Two eventual output conditions are distinguished from the point of view of residual heat removal from the spent fuel pool:

- spent fuel of previous lives is stored in the pool in order its activity and residual thermal performance lowers. The minimum level is +14.6 m at this stage.

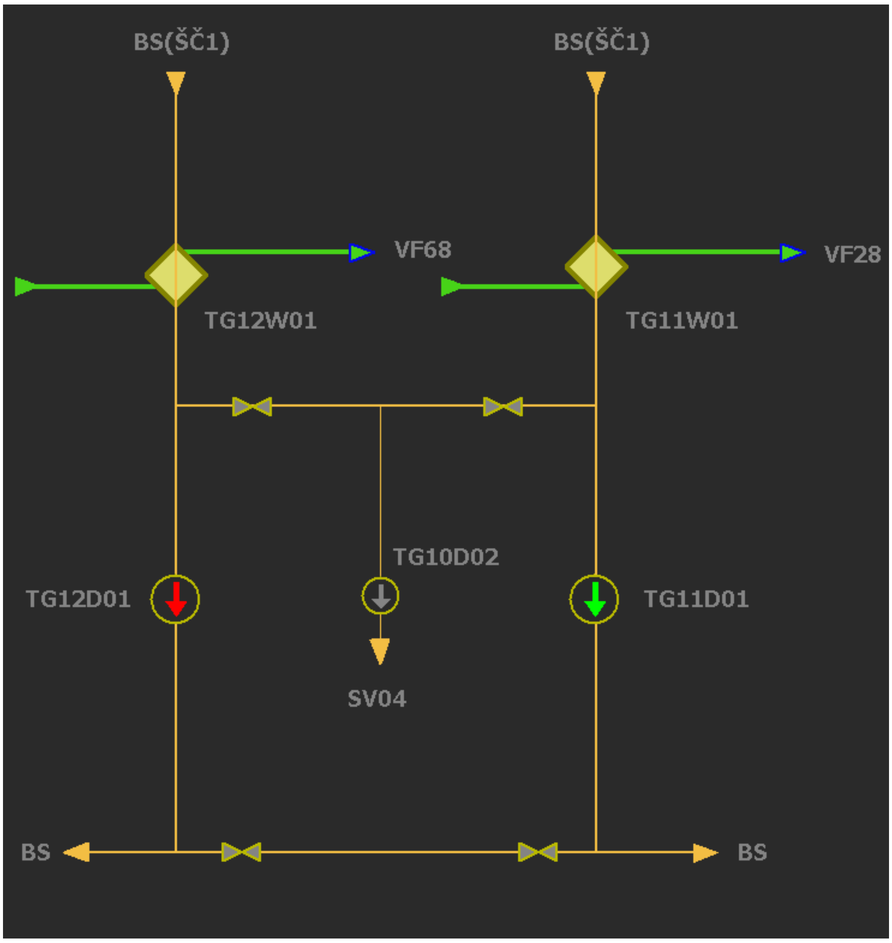
- during the extended unit outage aimed at general maintenance when all the fuel – spent fuel and partially spent fuel – is removed from the core. At this stage, the fuel is placed on two storage racks above each other; the minimum level is +18.5 m; the minimum level is +20.75 m when the fuel is exchanged and handled.



1.3.3.1 All existing heat removal means / chains from the spent fuel pools to the heat sink

The heat removal from the BSVP in both cases described above is provided by means of forced circulation of the cooling water provided by the BSVP cooling system marked TG. Circulation is provided by means of two pool cooling pumps and heat removal is executed in two exchangers cooled by the ESW. Simply, there are two separate and independent cooling circuits (redundancy 2x100%). In order to increase the reliability and operating ability of the system of heat removal from the BVP, these two circuits are, however, interconnected at the suction point of the pumps and also at the discharge point of the pumps, which enables to operatively combine the heat removal chain (pump, exchanger, ESW system). The pumps of the TG system are power supplied from the secured power supply of category II (DG) and the cooling system including the spent fuel pool itself is located in the reactor building of the HVB outside the containment.

If any of the combinations of the forced cooling circuit (TG pump, exchanger, ESW) is not able to provide heat removal from the pool, heat is removed by means of boiling and evaporation of the coolant from the pool into the reactor hall of the twin-unit of the power plant (there are pools of both units of the respective HVB in the reactor hall).



1.3.3.2 Respective information on lay out, physical protection, time constraints of use, power sources, and cooling of equipment

Operating systems

In common operation conditions, heat is removed by means of two circuits of the TG system belonging to the SSB and which is of the concept of 2 x 100%. Boric acid solution is the coolant. The cooling system is seismic resistant. The verifying calculations were carried out for the value of PGA = 0.1 g in compliance with the recommendations by IAEA. The cooling circuit capacity is sufficient for both output conditions for filling the BSVP.

Residual heat is removed by means of continuous coolant circulation through the coolers cooled by the essential service water from the BSVP. The ESW is the end residual heat sink in standard conditions. Loss of residual heat removal from the ESW to UHS is described in Chapter 5.2. If there is a complete loss of residual heat removal by means of the ESW system, it is possible to remove residual heat from the BSPV by means of boiling in the BSVP and adding boric acid solution eventually clean water on a long-term basis.

Resistance to extreme external events

The BSVP cooling system is located in the HVB in the reactor building and, thus, it is protected against adverse effects of the extreme external events. Its seismic resistance of value of 0.1 g was verified by calculations. The resistance of the ESW system to the extreme external events is dealt in Chapter 4 with.

If any of the combinations pump-exchanger-ESW is not able to remove residual heat from the fuel, it is possible to remove heat by means of discharging the coolant from the BSVP to the NT tanks of the SAOZ systems by means of heating the water in these tanks. Then there is no unwanted mixture of media because the NT SAOZ tanks are intended for BSVP make-up in refuelling modes - the solution in these tanks is used to increase the level in the BSVP in refuelling mode. For the purpose of discharging the coolant from the BSVP, it is possible to use the TG10D01 pump designed for this. This procedure would be followed until the temperature increases to 60℃in all SAOZ tanks. In this way it is possible to prolong the time to reach the boiling point in the pool. SD-9 operation procedure has been prepared if there is not sufficient cooling of the BSVP.

BSVP make-up from the VBK raceways that can be provided by means of the XL 10D01 pump of VBK raceway make-up is considered an alternative way to heat removal from the pool.

Another possible way of BSVP make-up is the option to use the TM13(14)D01TM pumps used for cleaning the BSVP coolant and to supply coolant from the NT SAOZ tanks to BSVP by means of these pumps. The TM pumps are power supplied from the system sources category II.

There is also a possibility to supply coolant by any of the VT or NT system SOAZ pumps directly to the reactor connected to the BSVP and from there to I.O or it is possible to provide coolant supply by means of discharging from the hydro-accumulators if the reactor is unsealed during refuelling.

If the coolant from all the SOAZ tanks and bubbler raceways is used, the coolant reserve is sufficient for compensating the loss by means of boiling the coolant in the BSVP for more than 8 days even if the fuel is arranged on two racks above each other.

The possibility to supply water evaporating into the reactor hall by means of mobile pump technology is the last resort of the way of BSVP cooling.

1.3.4 Heat removal from the reactor containment to the ultimate heat sink

The reactor and primary circuit of the reactor are closed in the hermetic area generally called containment. The containment is used to localize radioactive (RA) substances in case of coolant leakage accidents inside the containment. The tightness requirement to containment is defined as the maximum 13 % decrease in the weight of the dry air in 24 hours at overpressure of 150 kPA inside the containment. The tightness of the containments of the EDU units has been 1.8 to 6.3 % on a long-term basis. The containment is equipped with a spray and vacuum bubbler system to decrease the pressure and localize RA substances released in the containment during coolant leakage accidents.

The containment is accessible to no operating staff during the common reactor operation and its sealing is periodically checked. All pipelines going through the walls creating the border of the containment are equipped with closing mechanisms that, in case of an accident,:

- close the systems of the common operation of the JE if these valves were open before the “big accident” alert
- open the emergency and safety systems if these valves were closed before the "big accident” alert causing isolation of the containment

The functions of the containment are as follows:

- to limit the release of radioactive substances outside the accident localization
- to maintain radioactive radiation in the limits defined by the design
- to protect systems and elements whose failure can cause release of radioactive substances above the admissible limit

Containment consists of these basic construction elements:

- hermetic steel lining (protective tiling)
- ferro-concrete protective construction
- hermetic closures

Total free volume (geometric volume after deducting the volume of the large-volume technological equipment) of the containment per a unit (reactor) is 51,119 m^3 without adding water volume in the bubbler raceways.

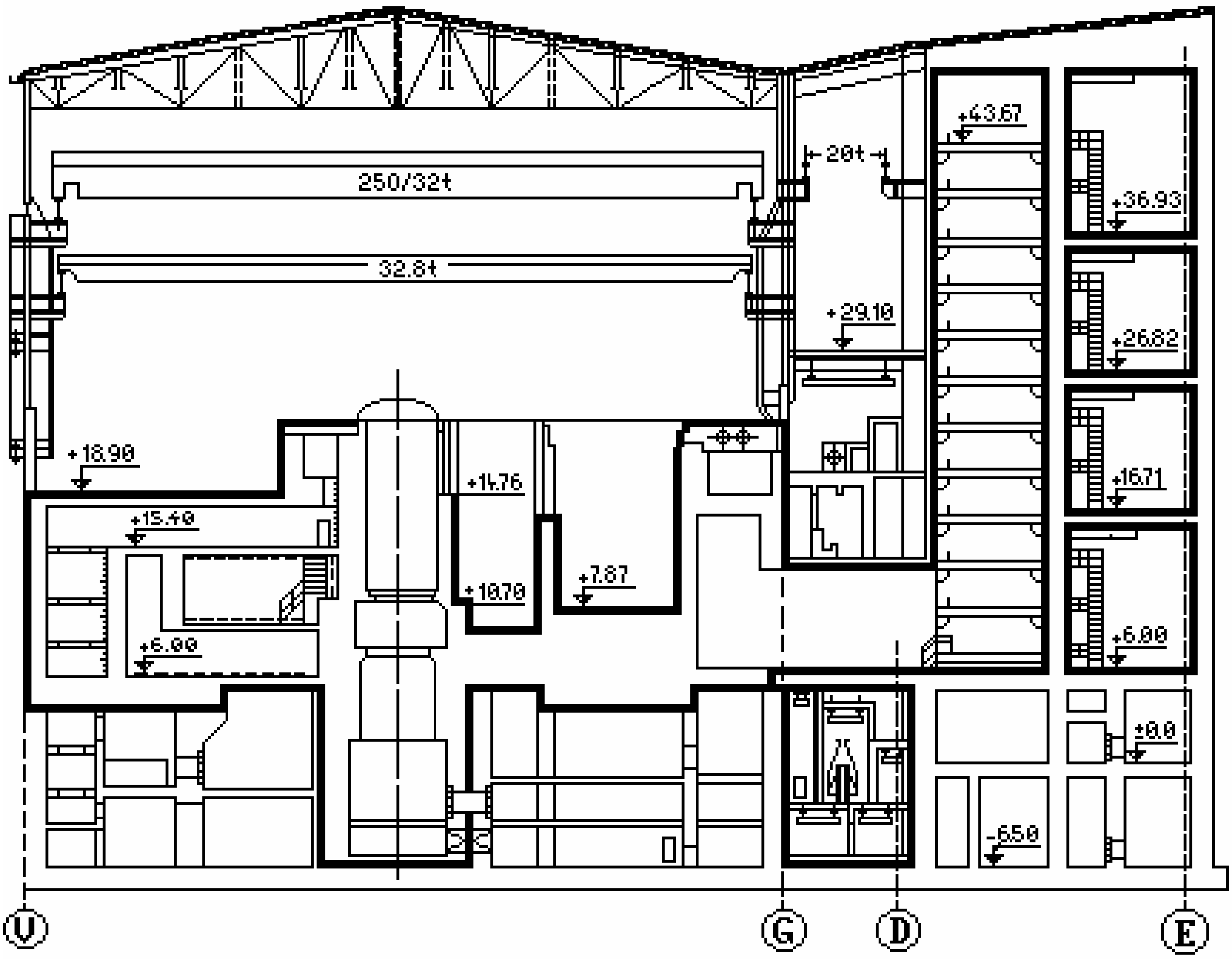


Fig. Side sectional view of the reactor hall of the JE with VVER-440/V213

1.3.4.1 All existing heat removal means / chains from the containment to the heat sink.

The system of heat removal from the protective envelope is divided into:

a) Ventilation recirculation systems (TL)

b) Spray system (TQ)

c) Vacuum bubbler system (XL)

a) Ventilation recirculation systems (TL)

The concept of heat removal from the containment by means of the ventilation systems is affected by two requirements:

- In order to minimize the values of the activity released to the environment through the vent stack, it is necessary as small as possible amount of air enters the vent stack (it is the activity carrier).

- In order to ensure suitable environment in the containment, it is necessary to eliminate a great amount of heat released from the primary circuit (during the common operation c. 1 MW).

Therefore the ventilation recirculation systems circulating only great amount of the air of the containment and ensuring its cooling and cleaning by means of special filters are used for removal of the released heat. Smaller ventilation systems (in and out TL40, TL70) removing a small amount of air from the containment through the filter system to the vent stack maintain the required under-pressure.

Recirculation ventilation systems are the basic systems of the heat removal from the protective envelope during the common and abnormal operation and partially in emergency conditions. They provide heat transfer to the ESW system by means of air circulation through the heat exchangers and they thus eliminate the burden released by the primary circuit technologies. They are involved in removing consequences of an eventual accident. The systems are located in the containment, i.e. area where (in case of accidents related to rupture of the sealing of the I.O, NV pipeline or hot steam) steam-gas mixture can spread. They are dimensioned and designed for operation in difficult conditions in terms of pressure, temperature and activity. The recirculation ventilation systems are located in the ventilation centre sharing the common air space with the steam generator boxes.

The reliability of the operation of the recirculation ventilation systems is provided in accordance with the operation importance with 50 - 200 % back-up units. The TL10, TL11, TL14 a TL13 systems are designed to remove heat from the protective envelope. These systems transfer the heat removed to the ESW or eventually to the cool water system VC transferring the heat to the circulation cooling water.

The TL10 recirculation system is designed to remove residual heat and moisture from the containment. It consists of three units; two operating and one back up. It is power supplied from the ZN of category 2.

The TL11 recirculation system is designed to remove residual heat and moisture from the bottom part of the reactor shaft and shaft of the reactor control and protection system. It consists of three units; one operating and two back up. It is power supplied from the ZN of category 2.

The TL14 recirculation system is designed to remove residual heat from the hall of electric drives of the TL10 and TL11 systems. It consists of three units; one operating and two back up. It is power supplied from the ZN of category 2.

The TL13 recirculation system is designed to remove residual heat from the hall of electric drives of the HNČ (HNČ boards). It consists of two units; one operating and one back up. It is power supplied from unsecured power supply.

b) Spray system (TQ)

The spray system is an active safety system designed to decrease pressure in the containment and remove heat releasing in the containment at the time of coolant leakage accidents. Released heat is removed by means of heat exchangers of the TQ system to the ESW. Every subsystem consists of a spray pump of flow of 380 to 520 m^3/ h and of overpressure at pressure discharge of 0.35 to 0.5 MPa, of an additional tank of c. 10 m^3 with N2H4 + KOH solution, of a water-stream pump, of a heat exchanger and of a spray nozzle system. Spraying cold water in the steam generator box by means of spray nozzles decreases the pressure. The system consists of three independent circuits; in order to manage all project-designed events it is sufficient to employ just one circuit. The suction of the TQ system is connected to the tank of the TH system (3x250 m^3 H_3BO_3 of concentration of 12g/kg) and when they are emptied the suction starts from the floor of the PG box through the coolers transferring heat to the ESW. It is power supplied from the ZN of category 2.

c) Vacuum bubbler system XL

The vacuum bubbler system is a passive safety system designed to decrease the initial increase in pressure in the containment during coolant leakage accidents. While it works, it absorbs a substantial part of the heat energy released from the primary circuit coolant leaked or medium leaked from the part of the secondary circuit located in the containment (feed water, steam). The vacuum bubbler system consists of a set of 12 storeys of raceways above each other filled with boric acid solution of concentration of 12 g/kg in which the released heat accumulates. The volume of a raceway is c. 114 m^3. The raceways create a water closure of little hydraulic resistance. The lower area of the water closure is connected with the containment; the area above the closure is connected by means of flap valve DN 500 with capture chambers and through flap valves DN 250 back to the containment. A triple of raceways above each other share a capture chamber– in total, there are 4 capture chambers of total volume of 17080 m^3. The capture chambers are designed to collect incondensable gases. The whole vacuum bubbler system is a part of the containment. If the I.O happens to be unsealed, a steam-air mixture going through the VBK raceways arises in the containment. The steam condensates in the water closure and the air go to the capture chamber where it is localized. If the specified pressure is reached in the containment, the DN 250 flap valves are automatically locked. The pressure is decreased by means of the spray system in the containment and the water is spilt from the raceways by means of the overpressure of the air mass above the level of the raceways through the flow screens to the floor of the steam generator box – it is passive spraying of the containment. Neither power supply nor SKŘ is required for the system as the system operates passively.

1.3.4.2 Respective information on lay out, physical protection, time constraints of use, power sources, and cooling of equipment

The spray system is located in the reactor hall inside the HZ and the discharging pipelines go to the containment. The pumps of the spray system are located on floor -6.50 m in emergency system rooms 001, 002, 003. The spray system is solved using the system redundancy 3x100% including all supportive systems (cooling, power supply, control and ventilation). The equipment is in the stand-by mode being ready to automatically start if an accident occurs in common operation of the unit at the I.O coolant temperature of 90 ℃. It is power supplied from the ZN of category 2. The systems are designed for long-term operation. Cooling of the TQ system rooms is provided by means of own ventilation system TL22.

The vacuum bubbler system including the capture chamber is located in the reactor hall in the containment. The raceways of the vacuum bubbler condenser are located in the shaft of accident localization that is connected to the PG box through a corridor of section of 77 m^2. The system is constantly ready to fulfil its design function during the unit operation. It is designed as a passive way without any need of power supply and control. Cooling is provided by the TL10 system.

1.3.5 AC power supply

The Dukovany Power Plant (EDU) consists of four reactor units VVER 440 MWe. The electrical system is designed so it works in accordance with the principle of defence in depth and it provides reliable power supply to all systems important in terms of nuclear safety in all operation, abnormal and emergency situations.

The principle of "Defence in Depth of Electrical Systems and Grid Interaction" (DIDELSYS) is applied to the electrical systems of the EDU in connection with the machine technology design and SKŘ instrumentation. The DIDELSYS application starts in the off-site grids of extra high voltage to which EDU is connected and continues through the systems of common power supply for own consumption to autonomous systems of secured power supply. Functions and ties of the individual levels of DIDELSYS are briefly described in the following table.

Level	Whole JE (DID)	Electrical systems of JE (DIDELSYS)	Robustness of levels
1	Preventing deviations from the normal operation	• Non-sensitivity to deviations U, f • Stability of energy transmission • Dynamic stability • Island operation	• Tie to robustness of technology, SKŘ and structure • Robustness of el. systems (independence, redundancy, diversity) • Protection • Regulation, automation • Quality • Testing function • Operation instructions • Trained staff
2	Identification and remedy of events, states and conditions of abnormal operation	• Regulating TG to VS • AZR to back-up power supply	
3	Steps (measures) leading to prevention of development of or managing emergency conditions by means of design means	• Design (safety) functions of systems of secured power supply: • BS (1,2,3), • SSB (4,5)	
4	Prevention and mitigation of consequences of extended design conditions	• Procedures to manage SBO • Measures to support mitigation of SA consequences • (AAC functions)	
5	Measures to protect in case of radiation accident	Support of emergency control centres	

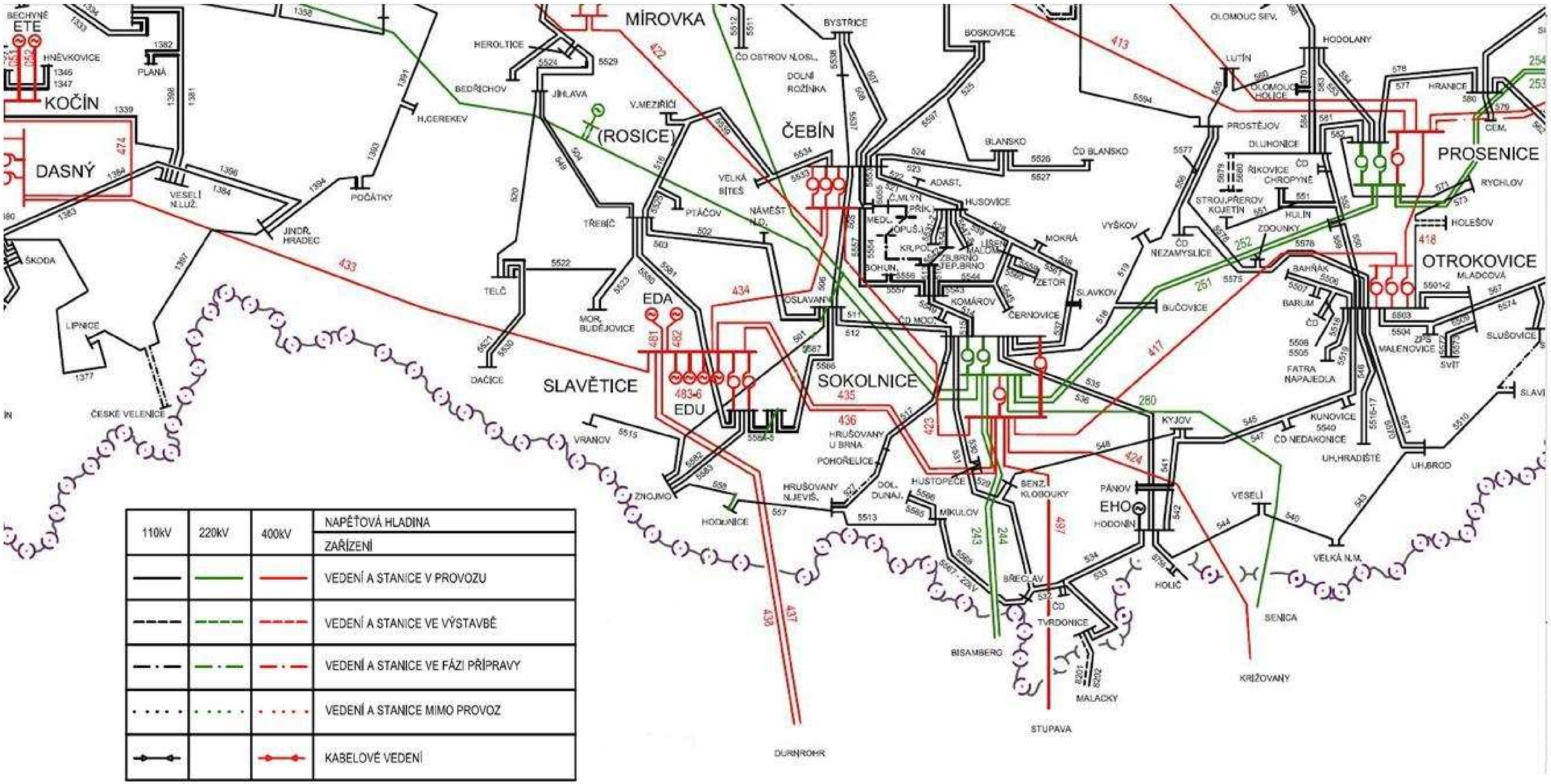
The resistance to the off-site and on-site events (failures) is based on the structure of DID and robustness of the individual levels.

The essential protective and control systems and the EDU and performance systems fulfilling the safety functions are power supplied from the redundant systems of secured power supply (SZN). Each unit of EDU is equipped with 3 independent safety SZN (marked 1, 2, 3) and other SZN classified as safety-related (marked 4.1 and 4.2). These SZN provide supportive safety functions such as secured power supply and they are employed in controlling the function of electrical appliances.

1.3.5.1 Electrical system off the EDU

The Dukovany nuclear power plant (EDU) is situated in the southeast part of the Czech Republic. The nominal output of the individual reactor units is being gradually increased from 440 MWe to 500 MWe within the project V261 "Utilization of the Design Basis Margins". Two pairs of reactor units are located in two HVB in terms of lay out. The system of leading-out the output and the system of operating power supply for own consumption is based on a unit design, the system of back up power supply from the grid of 110 kV is based on the twin-unit design. The output is led out to transmission grid of 400 kV (operated by ČEPS a.s.). The back up power supply is provided from the distribution grid of 110 kV (operated by E.ON).

The connection of the JE Dukovany to the electricity network of the Czech Republic (400 kV and 110 kV) described above can be seen in the following figure.



Mapa distribuční a přenosové soustavy 110 – 400 kV ČR, uzel JE Dukovany (2011)

1.3.5.1.1 Information on reliability of the external electrical system

No 400 kV and 110 kV grid failure showing any unsuitable function or reliability of the off-site electrical system or wrong response of the EDU to off-site grid failures has been recorded during the operation of the EDU (since the start-up of the individual units between 1985 and 1987 until today).

Several failures of the technology connecting the individual units to the grid resp. surrounding substations and lines of the transmission grid have occurred during the operation so far .You can find the summary in the table below. The most serious failure was a short circuit of the transfer bus bars of the Slavětice 400 kV substation in 1990 when, as a result, the coupling to the 400 kV and 110 kV grid of all 4 units was disconnected. On the other hand, the system failures in the transmission grid showed the ability of the EDU units to manage island mode of the grid operation in 2006.

12/1990	Bus bar short circuit at the Slavětice 400 kV substation due to wrong handling by the substation staff during its maintenance. All 4 units of the EDU lost coupling to the 400 kV and 110 kV grid from the Slavětice substation. Unit 1: TG not switched to own consumption, caused by the reactor protection system and DG successfully started. Then change to back up power supply from 110 kV. Unit 2: TG switched to own consumption. Unit 3: Maintenance outage; change to DG and back up power supply from 110 kV. Unit 4: TG switched to own consumption. The causes were identified, analysed in detail and removed.
2006 - 2008	Several failures of the electrical technology (chains of isolators of the 400 kV line, puncture in insulation of the control cables, surplus operation of distance protection systems of the 400 kV lines in the transmission grid). The EDU units manage to handle these events in accordance with the design; mostly it was about switching to own consumption. Causes of these failure events were identified and removed.
2006	The EDU units successfully managed the large system failures (loss of UCTE grid 11/2006, short circuit in the Sokolnice substation leading to creation of island grid in the surroundings of the EDU in 08/2006) and their resistance to voltage deviations and frequency and regulation abilities helped to stabilize the situation of the transmission grid.

The reliability of the EDU connection to the off-site grids and the resistance of the EDU to failures in this field are based on these characteristics:

- Unit arrangement of the system of output lead-out. It limits transfer and spread of failures between the units. It also limits transfer of failures between the units and the transmission grid in combination with the robust diagram of the Slavětice substation (2 breakers per a branch, sectional division of bus bars, selective protection system). Lead-out of the output of the EDU to the grid is designed in accordance with the reliability criterion N-2.

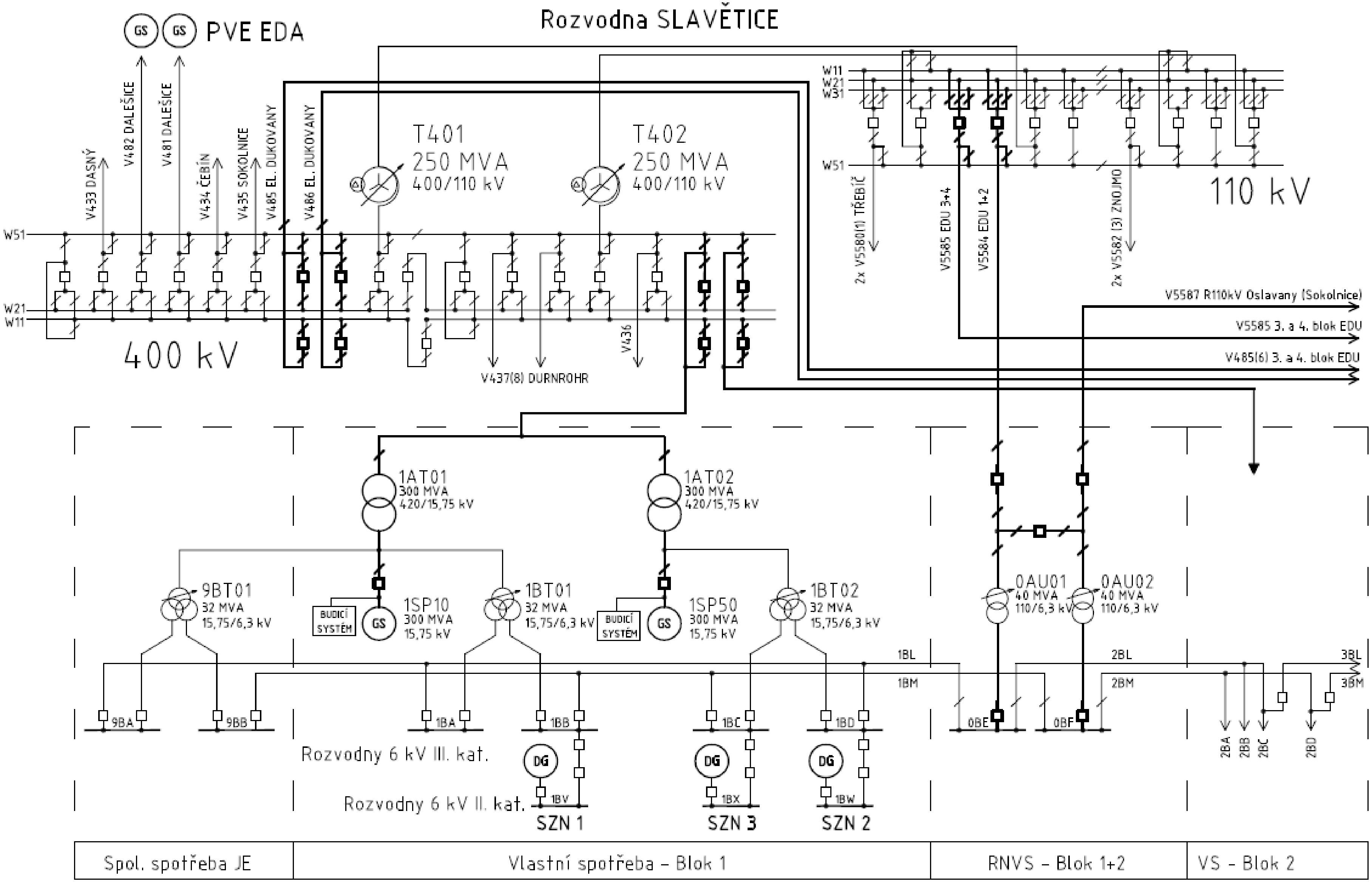
- Big functional and physical independence of the 400 kV output lead-out system (i.e. operating power supply for own consumption) and the system of back up power supply for own consumption (110 kV). A possibility to power supply the system of the back up power supply from various diverse sources in terms of geography and functions.
- Responses of the unit to failures and transients in the off-site grid are controlled by a set of regulations, automation and protection systems. These functions are inter-coordinated in order to provide mutual selectivity and if needed the unit proceeds at the individual levels of the defence in depth in a controlled way.
- The static stability of the transfer of the output to the grid. The EDU units are commonly placed in the system of automatic secondary regulation of voltage and reactive powers. This system provides stable voltage in the pilot junction of Slavětice.
- Stability of the turbine generator at short circuits in the system of the output lead-out (quick basic and back up protection systems that switch off the short circuit, effective regulation of the turbine and generator voltage). Stability has been analysed on a dynamic model of transmission grid. Considering the function of its regulation the turbine generator is stabile at operation of basic and back up protection systems (to 100 ms) and automation system failure of the 400 kV breakers.
- The ability of the unit operation in island operation of the transmission grid with big deviations in frequency and voltage (support of stability of the grid at system failures). The units can operate at full output in the range of 48.5 – 50.5 Hz. Unit operation limited in terms of time and performance is possible in the range of 47.5 - 52.5 Hz. The units are equipped with grid frequency protection systems that change the regulation of the unit output to "island regulation“ in 1° (±200 my). The unit regulates its output in this way in order to help stabilize the ratios of U and f in the island grid. If failing to stabilize the ratios and the frequency deviation becomes greater and exceeds the limit (47.9 Hz for a period of time of 1 s resp. 52.5 Hz for a period of time more than 10 s), the unit switches off the 2° frequency protection system fro m the grid and switches itself to own consumption.
- Disconnecting the unit from the 400 kV grid at the decrease at the 6 kV substations of own consumption is provided by means automation system that to 0.7 Un/ 1.5 s. After disconnecting from the grid, the unit switches to own consumption. If the switching-to is not successful, it changes to the power supply for own consumption from back up

sources (10 kV grid). If the change to the back up sources is not successful (110 kV grid) or if these sources are not available, it is power supplied from emergency sources (DG).

- The main unit substations are equipped with automation systems of substitution from the operating to back up power supply (110 kV). If the operating power supply for own consumption (e.g. activity of generator protection systems, unit transformers and other equipment of output lead-out or if the turbine generator is not successfully switched) the substations change to back up power supply. If the substitution is not successful, the systems of secured power supply are secluded and changed to emergency sources (DG, accumulator batteries).
- Regulation, automation and protection systems are power supplied from the sources provided by batteries. Their function is thus independent on decreases in voltage in the grid caused by failures. The principle of electro-magnetic compatibility ensuring operation of the systems in the respective electro-magnetic environment and at disturbance is applied to the whole design of the EDU.
- The EDU units are operated in accordance with the dispatching controlling of the transmission grid. The operator of the transmission grid knows the properties and operating limits of the EDU; this information is embodied in the PS Codex. Periodical maintenance of the equipment of the off-site grids (substations, lines, 400/110 kV transformations) and of the EDU technology is carried out in a mutually agreed way. In emergency situations in the electrical network (break up of the grid, SBO EDU etc.) the renewal of the power supply for own consumption of the EDU from the transmission grid is of the highest priority

1.3.5.1.2 Connecting the power plant to the electrical network

The basic electrical system of the EDU can be seen in Fig. 1.3.5-1. The system covers unit 1 and 2, the system of unit 3 and 4 is analogical. The system covers the electrical output lead-out to the 400 kV grid; the diagram of the back up power supply from the 110 kV grid and the diagram of own consumption to the level of the main 6 kV substations and connection of SZN 1, 2, 3.



Obr. 1.3.5-1 ZÁKLADNÍ ELEKTRICKÉ SCHÉMA EDU 1. a 2. blok

Each reactor unit is equipped with two turbine generators with increased capacity with generators of 300 MVA, 15.75 kV for the purpose of implementation of the project of utilisation of design basis margins. The output of each generator goes through the generator breaker and unit transformer (300 MVA, 420/15.75 kV) to the 400 kV electrical substation. The lead-outs of both generators are joined at this substation and they are led by means of a separate simple line to the 400 kV substation in Slavětice approximately 3 km far away from the EDU. There are only disconnections, device transformers and lightning arresters in the 400 kV electrical grid. The breakers of 400 kV of the EDU units are located in the Slavětice substation.

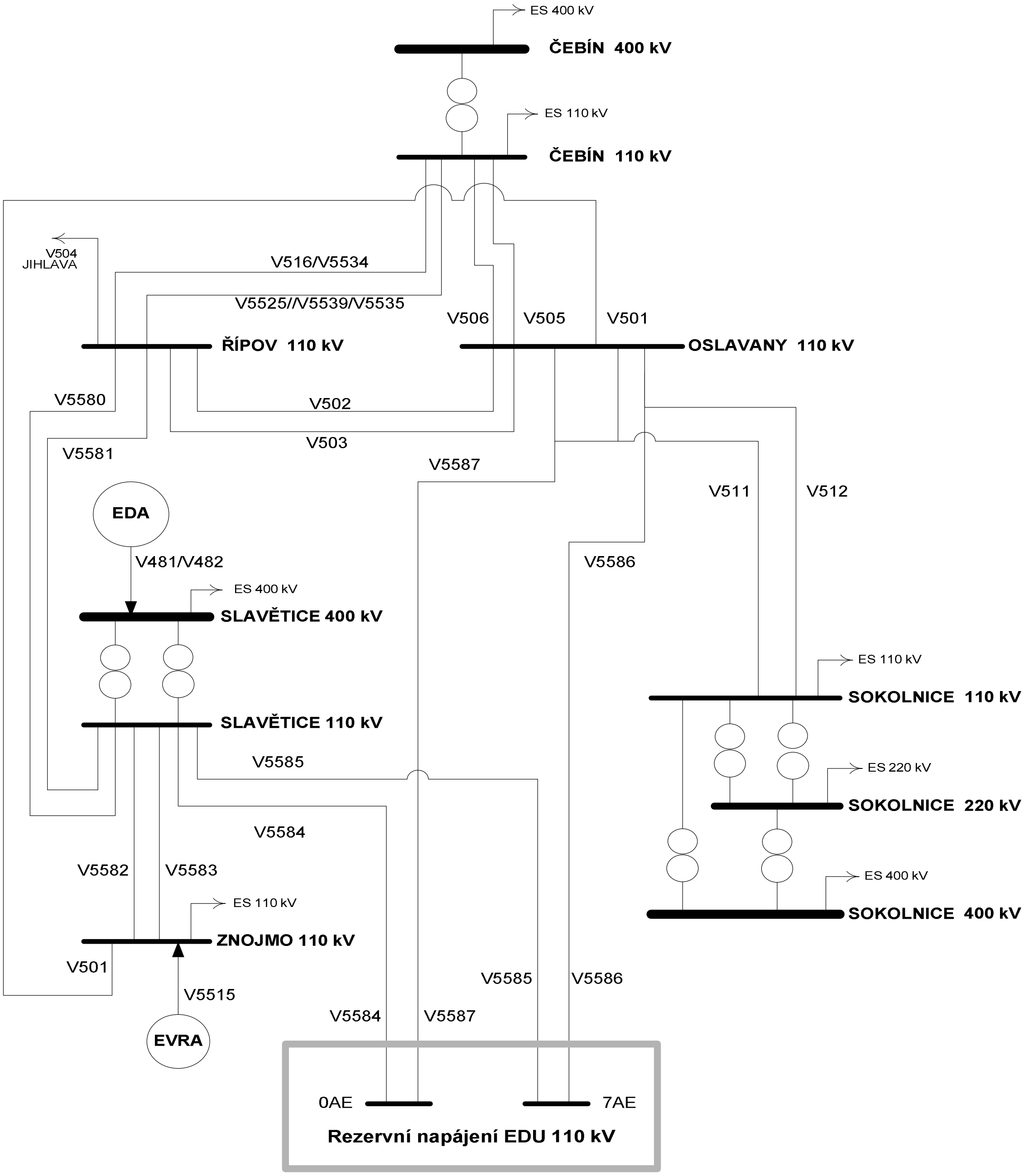
The 400 kV Slavětice substation is connected to the transmission grid by means of 6 lines; 4 lines distribute the output to the differently distant parts of the Czech Republic and 2 to Austria. Geographical diversity of 400 kV connection is created in this way. A part of the lines is simple; a part of the lines is double. The transmission grid of the Czech Republic as a whole is designed and operated in accordance with criterion N-1. However, the output lead-out from the Slavětice substation further to the grid is designed in accordance with the strictest criterion N-2. These requirements are laid down in the PS Codex.

The Slavětice 400 kV substation is of outdoor design with short circuit resistance of 50/125kA. It is of a system of two systems of bus bars and a transfer bus bar. Moreover, breakers longitudinally separate the main bus bars. The EDU units are connected based on the system of 2 breakers per a branch that has been chosen in compliance with the increased requirements to reliability and resistance of the EDU output lead-out against failures. Other lead-outs (to 400 kV lines to the transmission grid are of the diagram of 1 breaker per a branch). There are two 400/110 kV, 350 MVA transformers for power supply of the Slavětice 110 kV substation in the substation. Further, the units of the Dalešice pumped-storage hydroelectric plant (4 x 112.5 MW) being the external source of AAC for solving SBO of the EDU are connected by 400 kV two lines to the substation.

The back up power supply of the EDU is power supplied from the 110 kV grid. Each HVB is equipped with its back up source power supplied from 2 different substations in the 110 kV transmission grid. There is automatic substitution between both leads from the 110 kV grid. These sources are usually connected to the Slavětice and Sokolnice 110 kV substations where transformations of 400/110 kV resp. 220/110 kV are. There are other possibilities of power supply. Substantial variability and geographical diversity is provided in this way. This connection also provides back up of transformations of 400/ 110 kV and 220/ 110 as well as sources for back up power supply of the EDU at maintaining sufficient short circuit hardness of voltage for back up power supply for own consumption of the JE.

The Slavětice 110 kV substation serves mainly as the main source for back up power supply of the EDU units. Apart from this, the 110 kV transmission grid is power supplied from the substation in South Bohemia. The Slavětice 110 kV substation has a robust and flexible system with 3 systems of branches.

The Sokolnice substation that is the other usual source of back up power supply of the EDU is a significant junction of the transmission grid with transformations of 440/110 kV and 220/110 kV. It is c. 50 km far from the EDU.



The principal system of the sources of the back up power supply of the EDU.

The system of the 400 kV and 110 kV Slavětice, Sokolnice substations (and other substations in the surrounding of the EDU) and the way of their operation are chosen in order to limit the transfer of failures between the units of EDU and electrical network as much as possible.

Therefore the system of back up power supply of the EDU can be power supplied from various in terms of geography and direction diverse sources in the transmission network (400/110 kV transformations and 220/110 kV transformations Slavětice, Sokolnice and Čebín). After analyses of SBO were carried out the Dalešice hydroelectric plant has been chosen as the main off-site source of AAC and this function was in fact verified by the tests. The Dalešice power plant (capacity 4 x 112.5 MW) has the capability to „start from the dark“. The capability to start power supplying within 30 minutes (through the 400 kV line) or within 60 minutes (through the 110 kV line) was verified by the test. The test covered verification of the technical and organisational measures to manage SBO, functionality of the communication means, roles and procedures of key people if SBO occurs. The condition for using the VE Dalešice far away c. 6 km is the operation able state of the 400 kV and 110 kV substations and lines in the supply route.

The capability to start power supplying within 60 minutes (through the 110 kV line) from the Vranov hydroelectric plant was verified by the test.

1.3.5.2 Electrical system in the EDU

1.3.5.2.1 Main supply sources and grids

Power supply for the electrical components of own consumption is divided in to more substations, supplying systems and sources that back up (based either on substitution or redundant principle). Consequences of failures of these systems are limited to the reactor and unit operation in this way.

Electrical components are divided into groups according to their importance and based on this they are power supplied from sources and grids of corresponding category of secured power supply. The importance of the component covers the criterion of (safety) function of the component and admissible time of power supply interruption. The function of the component is classified in accordance with the IAEA standards as safety (BS), safety-related (SSB) and unimportant in terms of safety (SNB). The own consumption of each of the units of the JE Dukovany has available:

- operating sources i.e. branch transformers ANT with power supply regulation OLTC (power supplied from turbo generators of 300 MVA (259 MVA for units before reconstruction in progress) and/ or from the 400 kV grid). The operating sources are of purely unit character.
- back up sources i.e. back up transformers AST with voltage regulation OLTC (power supplied from 110 kV grid). There are two back up transformers per every reactor twin-

unit (HVB) and they can be backed up by AST of the neighbouring HVB. The back up transformers are able to provide trip of a reactor unit at loss of its operating power supply at preliminary load by components of the other reactor unit.

- emergency sources that power supply the systems of secured power supply (SZN). Emergency sources consist of diesel generators, accumulator batteries and aggregates of uninterrupted power supply (rectifiers, bridge inventers). They are installed in the premises of the EDU, dimensioned in accordance with the requirements to supplied loads and their function/ ability is not dependant on the state of the operating and back up sources of the off-site grid, too. Each of the EDU units is equipped with 3 redundant SZN (marked 1, 2, 3) classified as BS (each of them is a supportive system for its division BS). SZN 4.1 and 4.2 for components of SSB and SNB power supply are power supplied from SZN 1 and 2.

Components unimportant in terms of nuclear safety (SNB ensuring unit operation, el. energy, …) are power supplied from the operating sources. If operating power supply is lost (ANT transformers), there is automatic change to power supply to back up sources (AST transformers). There are 4 main unit 6 kV substations (BA, BB, BC, BD) equipped with AZR for back up power supply in each unit.

The components important in terms of nuclear safety (BS and SSB) are power supplied from the systems of secured power supply (SZN). SZN consist of grids of secured power supply and emergency sources. SZN are commonly power supplied from operating or back up sources (by means of substations BA, BB, …). If this power supply is lost, the respective SZN disconnects from the grid of the common power supply and changes to power supply from the emergency sources. Accumulator batteries provide uninterrupted power supply of sensitive components.

Disconnection of SZN and start up of DG is started from the loss of power supply ($U<0.25Un$ for a period of 2.5 s or 0.7 Un/6 s or decrease in frequency of 47 hz/6 s). Deviations in frequency are solved by means of grid frequency protection that evaluates the decrease in frequency in the 400 kV grid. The project analyses and tests confirmed selectivity of this set up against the AZR function from the operating to back up sources and function of other protection and automation systems.

DG are automatically started and automatically gradually loaded by means of fixed specified ELS programme from occurrence of these initiating conditions. DG are ready to be loaded within 10 s from the start order in compliance with the safety requirements. The DG and their loading automation system function are verified by periodical tests on a regular basis.

1.3.5.2.2 Arrangement and distribution of sources and grids

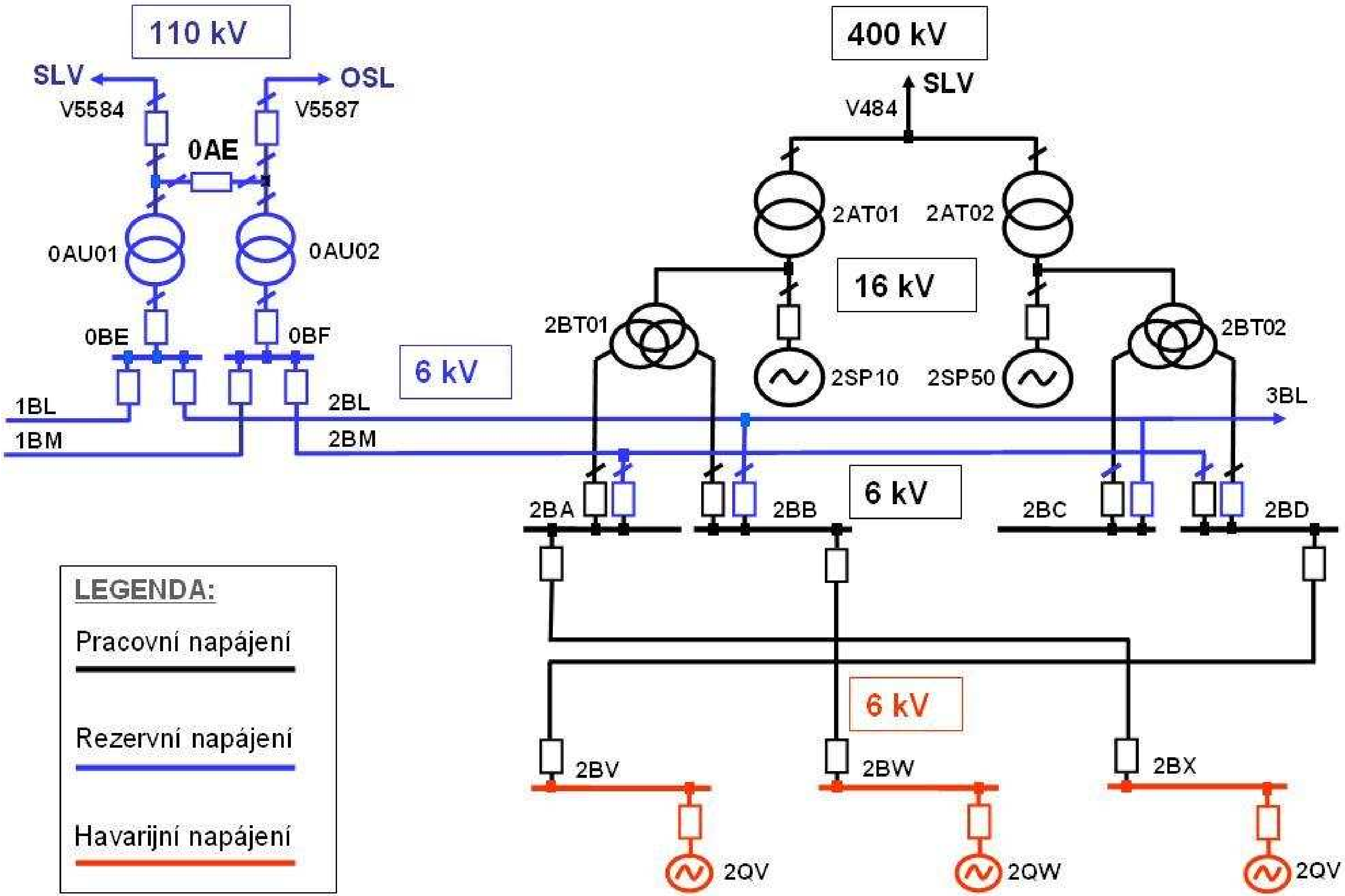
There is a place of unit (GT), branch (ANT) and back up (AST) transformers in front of the turbine hall of the unit. The places are separated from each other physically, electrically and in terms of fire protection.

The operating sources of own consumption (2 branch transformers ANT, each of capacity of 32/16/16 MVA) are power supplied from the branch of the main generator. ANT power supplies the main unit 6 kV substations (BA, BB, BC, BD) that are located in the intermediate machine hall (longitudinal and cross). There are 6/0.4 kV lowering transformers and 0.4 kV leads for power supplying of the reactor hall, turbine hall and secondary circuit in this hall.

The back up sources for own consumption (there are 2 back up transformers AST, each of capacity of 40 MVA, per each HVB) is power supplied from 110 kV grid. The back up sources of HVB1 and HVB2 can back up each other by means of 6 kV connections. AST are sources of back up power supply of unit 6 kV substations BA, BB, BC, BD.

The 6 kV engines (HCČ, feeders, coolers, ….), 6 kV substations of secured power supply (BV, BW, BX systems 1, 2, 3) and 6/0.4 kV transformers for components of the turbine and reactor halls are power supplied from the unit 6 kV substations, too. These distribution devices are located in the longitudinal and cross intermediate machine hall.

Further, 6 kV drives and 0.4 kV leads located in the outer buildings are power supplied from the unit 6 kV substations. See the following figure:



Systém napájení VS – 2. RB EDU

1.3.5.3 Normal design systems of secured power supply

1.3.5.3.1 System, basic parameters and properties

1.3.5.3.1.1 SZN for power supply of safety systems (BS):

Safety systems (BS) are arranged in 3 divisions of safety systems (3x100%) in each of the EDU units. In accordance with this concept, a SZN (marked 1, 2, 3) serving as a supportive safety system for power supply of components of this division is created in each division (marked 1, 2, 3).

1.3.5.3.2 Redundancy, separation of redundant sources by structures or distance

To secure the necessary degree of redundancy, these SZN are independent and separated from each other in terms of their layout (structurally as well as from the point of view of fire protection), the electrical arrangement and the control system. SZN 1, 2, 3 are seismic resistant. Each of the SZN is equipped with own emergency sources (DG, accu batteries) and electrical leads. The SZN 4.1 and SZN 4.2 systems of lower classification in terms of safety (SSB, eventually SNB) for which high level of reliability and redundancy is considered are power supplied from SZN 1, 2. These systems must not, however, lower the safety function of the BS systems.

Each of the SZN 1, 2, 3 consists of these main devices:

- Emergency DG of 6.3 kV, 2.8 MW. Diesel generators (QV, QW, QX) have their own fuel oil tanks that are dimensioned for operation at full load for at least 144 hours without any fuel oil refill (for a longer period of time in reality because the load is lower). Moreover, fuel oil can be refilled on a interim basis
- 6 kV substation of secured power supply.
- The 0.4 kV leads and 6/0.4 kV lowering transformers.
- Rectifiers, accu batteries, inverters for power supply of sensitive components requiring quality and uninterrupted power supply.

The BS1, 2, 3 divisions and their SZN 1, 2, 3 are backed up as a whole (concept 3x100%). Functionality of the remaining two divisions will not be disturbed by a simple failure of one of the SZN 1, 2, 3 due to the principle of independence and separation from each other.

DG are emergency power source for components allowing interruption of power supply for a certain period of time (dozens of seconds to minutes). DG are started automatically at loss of power supply from the 6 kV substation of the secured power supply of its SZN. At the same time this substation is disconnected from the common power supply by means of switching off two in a series arranged sectional breakers. Loading of DG and work of SZN and components are of the highest priority controlled by the sequencer (ELS) in accordance with the fixed defined programmes. The ELS also protects DG against over-loading due to other accident development and following actions by the operator.

SZN 1, 2, 3 (including DG) consist of seismic resistant device located in buildings where seismic resistance improvement is carried out.

The buildings of DS stations for SZN 1, 2, 3 are robust ferro-concrete buildings located behind the reactor halls.

The main electrical devices of each of the SZN 1, 2, 3 (substations of 6 kV, 0.4 kV, batteries …) are located in the ross and longitudinal intermediate machine hall and thus protected against external risks.

Cable routes of SZN1, 2, 3 are independent of each other. Functional, spatial and fire protection independence (90 minutes) of these SZN and respective BS divisions is provided in this way. Cables are segregated according to the functional and voltage groups in the cable routes.

All auxiliary systems of the engine 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 of the DG is equipped with own lead and source of own consumption including own battery. 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 fuel oil quality is checked on a regular basis.

It is also possible to carry out tests of function of the DG and SZN during the unit operation (test of start-up of DG and taking over the loads of ELS function after intentional switch off of sequential breakers and simulation of real power supply) due to the concept (3x100%) of the safety system divisions and SZN 1, 2, 3.

1.3.5.3.2.1 SZN for power supply of the safety-related systems (SSB):

In each unit, SZN 4 is created for power supply of the part of the systems related to nuclear safety (SSB) and for power supply of the systems irrelevant from the point of view of nuclear safety (SNB) which, however, provide general safety of persons and expensive equipment such as turbine generator this SZN is designed as two sub-systems (4.1, 4.2) mutually redundant based on the 100+100% principle. Each of the sub-systems is equipped with own accu battery, rectifier and invertor. The SZN 4 devices are seismic-resistant.

1.3.5.3.3 Time constraints for availability of these sources and external measures to extend time of use

1.3.5.3.3.1 Fuel oil management:

The reserve of fuel oil in the operating tank for each DG is sufficient for a period of at least 6 hours (4.5 m^3 of fuel, consumption of 0.7 m^3/h). For each DG, one, interconnected pair of storage tanks has been determined, which contains a minimum reserve of 110 m^3 of fuel. Re-pumping of fuel oil from the storage tanks into the operating tank proceeds automatically in case of drop of level on the operating tanks. The pumps to transport fuel are power supplied from the respective DG. Total fuel oil reserve of 114.5 m^3 is sufficient for operation of a DG for a period of at least 144 hours (in fact c. 160 h), i.e. which means 6 to 7 days without the necessity of external fuel supply.

Note: Other fuel for DG could be obtained by dispatching from the storage tanks of the other DGs (that are for instance out of operation) the so-called re-expedition pumps under condition of obtaining electrical power supply for their operation (power supply of pumps is one of the unsecured systems). While considering a long-term operation of always just one DG on each unit, putting the dispatching pumps into service would mean to have fuel for a period of 18 to 21 days at the disposal without the necessity to supply EDU with fuel oil.

1.3.5.4 Alternative systems of secured power supply permanently installed in the JE/PP

Measures and procedures for handling of and renewal from the total loss of alternating power supplies of VS (Station Blackout, hereinafter referred to as “SBO”) are designed by virtue of keeping the units safe for JE Dukovany in the operational guidelines. The basic method of solving SBO is close to US NRC RG 1.155. The SBO is to be resolved from the point of view of a single unit in case the whole plant is affected by LOOP.

The definition of the Station Blackout state (SBO) on one unit of EDU: SBO is an accident on a unit of the JE Dukovany characterized by loss of all operating, back up and emergency sources of alternating power supply of a unit - for the unit at the output e.g. after the electrical network has broken, neither of the two turbo-generators did not switched to own consumption and no power supply from any of the three diesel generators of a unit.

1.3.5.4.1 All diverse sources that can be used

The following power supply sources are considered for power supply renewal in case of an SBO inside the EDU:

- use of emergency DG of other unit in which at least 2 DG are in operation. Alternative procedures of operation handlings when expressway of back up bus bars of 6 kV going along the whole length of all 4 units of the EDU used for connection are in the guidelines.

- a procedure of renewal through the supply routes of the crane route of the reactor hall is prepared for renewal of power supply of the pumps of BSVP cooling (TG).

1.3.5.5 Other sources and systems for solving emergency situations

1.3.5.5.1 Potential dedicated connections to neighbouring units or to nearby other power plants:

The Dukovany Power Plant has a very robust system of the so-called back up power supply at its disposal – see Chap. 1.3.5.1 offering a wide range of possibilities of power supply of important components resp. renewal of their power supply from neighbouring sources after SBO:

- Basic (verified, in time of up to 30 resp. 60 minutes) from the pump-storage VE Dalešice according to the current situation of lines 400 or 110 kV – see 1.3.5.1
- From VE Vranov through lines of 110 kV through Znojmo substation and according to the current situation either through the Slavětice 110 kV substation or through the Oslavany substation – it is considered especially in situations if the Slavětice substation including the lines of 110 and 400 kV would be put out of operation between this substation and the JE Dukovany (also verified within 60 minutes – 1.3.5.1) due to local extreme natural phenomena
- Power supply from any source connected to the Sokolnice (400, 220 or 110 kV) or Čebín (400 and 110 kV) substation through the Oslavany substation – suitable especially if local breakage of the 400 and 110 kV grids that would put both voltage grids in the Slavětice substation out of operation.
- Power supply from any source connected to the Řípov or Znojmo (incl. VE Vranov) substations through R 110 kV Slavětice – suitable especially if R 110 Oslavany is out of operation e.g. due to local extreme natural phenomena.

Putting into operation incl. eventual repair of any (the least affected) of the above mentioned routes will take hours to dozens of hours, which is comparable to the time needed to activate (incl. transfer) mobile or other alternative power supply sources.

1.3.5.5.2 Power supply from mobile sources

Use of mobile sources has not been considered in the design of the EDU.

1.3.5.5.3 Information on each power source:

The pump-storage VE Dalešice is located c. 6 km far way from the EDU and it can be connected by means of the 400 kV, resp. 400 kV and 110 kV grid through the Slavětice substation. If the possibility to earmark the 400kV or 110kV power supply lines for the EDU remains available, the VS power supply of the selected units would be provided primarily from the nearby Dalešice (4 x 112.5 MW) and Vranov (3 x 6.3 MW) hydro power plants in compliance with procedures specified in AOPs.

However, the precondition is the possibility to communicate with respective off-site dispatching and workplaces of the hydro power plants. These methods of voltage renewal from the off-site power supply units are conditioned by the availability of the necessary segments of the 400kV and 110kV lines, which may not be available.

Renewal of power supply from the pump-storage VE Dalešice (4 x 112.5 MW) resp. from the VE Vranov (3 x 6.3 MW) was tested (in 2004, resp. 2010) with satisfying result.

1.3.5.5.4 Preparedness to take the source in use:

If SBO occurs in the unit in hot state, SI EDU shall declare the state of DESTITUTION that in accordance to the Codex of Transmission Grid of the Czech Republic (Act No. 458/2000 Coll., Rules for operating Transmission Grid) lays down the necessity to supply energy from off-site grid to the affected unit within one hour. If SBO occurs on the unit in semi-hot state, the state of DANGER shall be imposed with the necessity to supply power from the off-site grid to the affected unit within 2 hours.

Renewal of power supply to the safety-relevant equipment (on condition that the respective off-site routes and switchgears are available) is provided by means of the following variants:

- Renewal of voltage from the Slavětice or Oslavany switchyards through the 110kV line.
- Renewal of voltage from the 400kV system through the 110kV line.
- Renewal of voltage from the 400kV system through the 400 kV unit line.
- Renewal of voltage from EDA through the 110kV line.
- Renewal of voltage from EDA through the 400kV unit line (tested in 2004).
- Renewal of voltage from EDU unit through the 110kV line.
- Renewal of voltage from EDU unit through the 400 kV unit line.
- Renewal of voltage from the Vranov hydro power plant through the 110kV unit line (tested in 2010).

All the aforementioned methods of power supply renewal from the off-site power supply units are conditioned by the operation ability of the 400kV and 110kV lines, which may not be available.

1.3.6 Battery sources for power supply of DC

Each of the SZN described in Chapter 1.3.5.3.1 is equipped with sources and distribution lines providing uninterrupted power supply of sensitive components. Lead batteries of 220 V are an emergency source. Checks and technical adjustments of set-up and coordination of protective and monitoring systems providing the robust resistance against failure and transfer process in the AC grid were carried out in all SZN following the events in the JE Forsmark. The loads are power supplied and batteries are re-charged via rectifiers from source of common power supply in the common operation. If operating and back up sources fail, power supply of the rectifiers is taken over by the emergency diesel generators.

The suggested rectifiers are able to provide re-charging of batteries in less 8 h.

1.3.6.1 Description of battery sources

Systems consisting of 2 thyristor rectifiers (220 V, In=800 A, current limit set up to 600 A), accu battery (220 V, 1500 Ah) and two transistor invertors (220/380 V AC, 160 kVA) are installed to SZN 1, 2, 3. These systems serve for power supply of the most important controlling, monitoring and protective systems and valves of its BS division. The important appliance is emergency lightning of the area, too.

Except for this, there are two subsystems 48V (100 + 100 %) power supplying the system of protection and control classified as BS (reactor protection, ELS automation etc.) on each SZN 1, 2, 3. Each of the subsystems is equipped with a 243 Ah battery.

The discharging period of 2 hours is defined by the design for all batteries of SZN 1, 2, 3.

SZN 4 contains two subsystems (4.1, 4.2) power supplied from SZN 1 and 2. Each of the subsystems contains a thyristor rectifier (220V, 800A), accu battery (220 V, 2000 Ah) and invertor (220/380 V AC, 150 kVA). There is a connection enabling mutual redundancy 100% + 100% between the distributer 220 V DC. These systems are classified as SSB and power supply components of the control system classified as SSB or SNB further assuring drive of turbine generator. Subsystems 4.1 and 4.2 mutually redundant (100% + 100%); the components are equipped with leads from both subsystems. Expect for this, SZN 4.1 and 4.2 are sources of UPS for power supplying less important control systems and diagnostics.

SZN 5 contains a battery in the system of drives of the controlling rods of the reactor (220 V, 600 Ah), it stabilizes this system at short-term decreases in voltage that can occur in the transmission grid or in the grid for own consumption.

Other battery systems of safety importance are in diesel generator stations. They consist of rectifiers and batteries of 24 V. They are power supplied from the distribution lines of own consumption of their DG. They power supply the controlling systems and systems of DG protection; the time of discharging at this load exceeds 8 hours. They are classified as well as DG (i.e. DG to SZN 1, 2, 3 as BS).

In the design of the EDU mainly in connection with the renewal of SKŘ, battery dimensioning was based on the requirement by IAEA NS-G-1.8:2004 i.e. time of discharging for at least 2 hours.

Discharging time of accu batteries in accordance with the design

SZN	Battery marking	Battery characteristics	Discharging time [hours]
1,2,3 220V	EE01,02,03	105 cells Vb2415, 1500Ah	**2**
4.1 + 4.2	EE04, EE14	105 cells Vb2420, 2000Ah	**2** *1)*
1,2,3 48V	EE5x,6x	24 cells Vb6159, 243Ah	**4** *1)*

x ... 1,2,3 ... acc. to the competence to SZN

Discharging time of accu batteries in the above-design mode SBO (defined in Chapter 1.3.5.4)

SZN	Battery marking	Battery characteristics	Discharging time [hours]	
			a) Without load reduction	**b) With min. reduction load**
1,2,3 220V	EE01,02,03	105 cells Vb2415, 1500Ah	**4 - 8**	**8 - 10**
4.1 + 4.2	EE04, EE14	105 cells Vb2420, 2000Ah	**5,5 - 12**	**6,5 - 12**
1,2,3 48V	EE5x,6x	24 cells Vb6159, 243Ah	**6**	**7**

x ... 1,2,3 ... acc. to the competence to SZN

Accu batteries are not critical from the point of view of over-design mode of SBO (defined in Chapter 1.3.5.4) by means of method of connecting the AAC source:

a) can be connected to the AAC source providing power supply of the load and re-charging of the accu batteries by means of rectifiers.

b) Discharging of accu batteries can be extended only by means of minimum reduction of their load up to the range of 6 - 8 hours when the connection of the AAC source is considered feasible and needed from the point of technology, too.

1.3.6.2 Power supplied components

Main types of power supplied components are:

- power supply of the most important controlling, monitoring and protective systems (RTS, ESFAS, PAMS) and valves of the BS division (PSA, OVKO, etc.)
- the important component is also emergency lightning of the area of the BS division in question (classified as SNB).

1.3.6.3 Arrangement and distribution of sources and grids

Battery systems SZN 1, 2, 3 and 4 and respective rectifiers and invertors are located in the cross and longitudinal intermediate machine hall. This is a seismic-resistant device located in the seismic-resistant rooms.

Other data see Chapter 1.3.5.3.

1.3.6.4 Alternative options of battery charging

Battery charging from the AAC sources listed in Chapters 1.3.5.3 and 1.3.5.4 is considered for the SBO mode.

In the current design as at June 30, 2011, it was not considered to charge batteries in any other way e.g. by means of mobile DG. However, there is this possibility from the technical point.

1.4 Significant differences between units

Utilization of the design basis margins of the units

In 2009, on the third unit and in 2010, on the 4th unit, a project of Utilization of the design basis margins of the EDU units (VPR) was implemented; within the framework of the project, the nominal thermal output of the reactor was increased from the original value of 1,375 MWt to the value of 1,444 MWt. The VPR project included refurbishment of the VT components of the turbines of the respective units. Implementation on the first unit is planned in 2011 and on the second unit in 2012.

Elimination of total loss of coolant in case of LOCA

In 2009, on the third unit and in 2010, on the fourth unit, a modification of the ventilation systems piping geometry was implemented on the supply system to the reactor cavity in the connecting corridor. This concerns two liquid seals with the lower edge at +8.1 m installed in the upper part with rupture discs (DN200), with opening pressure of 50 kPa on the side of the corridor against forcing back liquid in the flooded pipeline. The purpose of the modification is to prevent the coolant getting into the spaces of the ventilation centre when the aforementioned value of the level is reached in the reactor cavity´s room. Implementation on the first unit is planned in 2011 and on the second unit in 2012.

Properties of TNR

The current PSA is a higher CDF value on the first unit. This is due to the fact that TNR on the first unit has worse properties as far as brittle fracture is concerned. Which means that worse results are achieved upon cold pressurization and worse results are achieved by LOCA sequence when pressure and temperature shocks might occur.

1.5 Scope and main results of Probabilistic Safety Assessments

The first analyses of PSA for the JE Dukovany, units 1 through 4 were performed during 1993 - 1996. The project of the EDU analyses covered the analysis of level 1 (Level 1 PSA) for power conditions and a limited set of on-site initiating events. These analyses were then extended gradually by analyses of other types of risks as well as by non-power conditions, including outages, risks of on-site fires and floods, falls of heavy loads, risks of off-site events. The PSA analysis Level 2 was prepared in 1998 and then updated first in 2002 and subsequently in 2006 as well.

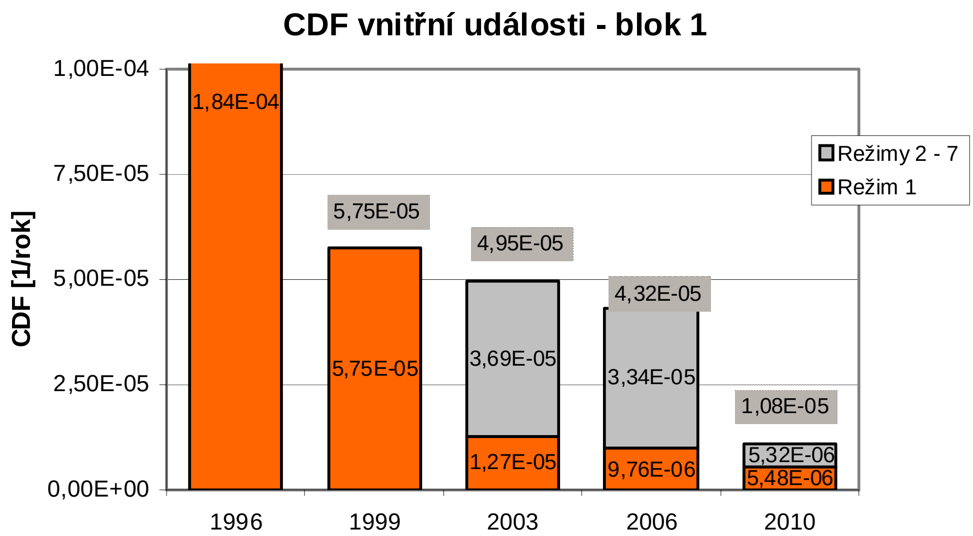
The original probabilistic models were further updated on a continuous basis to be able to reflect the actual condition of the units´ design after all the gradually implemented safety-related improvements. The update of the models included analysis of fire risks, risks of floods and the update of models of PSA Level 2. The PSA analysis Level 2 at the present time includes power operation; for non-power operation and outages, the PSA analysis Level 2 is being prepared.

The PSA for the Dukovany JE was the subject of the inspection mission of IAEA IPERS in 1998 (PSA Level 1, on-site initiating events, fires, floods). An independent assessment mentioned by PSA was performed as well (within the scope of PSA Level 1 for on-site events, for power as well as non-power operation and PSA Level 2), after it had been initiated by SÚJB and it was performed by the Austrian ENCONET Consulting company in 2005 and an annual verification inspection and assessment of PSA is performed on an annual basis by SÚJB.

Update of the PSA probabilistic models is performed regularly within the framework by the Living PSA concept adopted by the operator also as a consequence of requirements imposed by the nuclear safety supervision authority (SÚJB) with respect to regular update of the EDU PSA models so that they results reflected the actual condition of JE and met the basic requirement of the usability thereof in case of the risk-informed applications.

The overview of development of safety expressed in the form of calculated frequency of the AZ damage (CDF) and frequency of early large releases of radioactive media (LERF) for power operation as well as non-power operation and outages is provided in the following diagram:

Diagram: Development of results of PSA Level 1 (CDF) for power operation EDU (R1) and for non-power operation/outages (R2-R7), on-site initiating events + fall of aircraft:



(Legend: CDF on-site events – unit 1, režim = mode)

CDF refers to value of the average annual frequency of fuel damage in AZ for power and non-power operation of JE

CDF (R1-R7) for the remaining EDU units je lower and amounts to in total 8.76 ÷ 8.81E-06/year.

Of the total value of the frequency of damage to AZ (CDF), the contribution of the category of early large releases of radioactive media (LERF) amounts to approx. 23.5 %.

Fully functional probabilistic models for monitoring of the risk in real time, the so-called Safety Monitor, have been prepared since 2000-2003, and since 2005 they have been operated and updated on a standard basis as well. It is used for identification of risk configurations of all units in the course of outages as well as for monitoring of risks profiles in real time both during operation and during outages of the individual units. It is used for the assessment of the operation related risk for the purpose of the risk-informed applications.

According to the current state of knowledge, in consideration of off-site design basis events, the following conclusions of the PSA analyses Level 1 are applicable:

- The current value of the contribution of a seismic event to the overall CDF risk is not available for the time being, the seismic risk analysis is being prepared. The prerequisite is a low contribution to the total risk with respect to the SSC qualification requirement for seismic resistance min. 0.1 g and low frequency of occurrence of earthquake with the respective intensity in the location.
- The contribution of an off-site initiating event caused by human activity to the risk is negligible (the value of contribution to CDF is approx. 1.0E-7/year)
- The contribution of accident sequences resulting in SBO due to on-site causes amounts to approx. E-7/year.

2 Earthquakes

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 the region of Central Europe and in the territory of the Czech Republic, there are no tectonic structures which might result in extremely severe earthquakes comparable to catastrophic earthquake in Japan that occurred on 11 March 2011.

The level of seismic risk to the location is determined by the value of maximum design basis earthquake with the time between occurrences 1 x 10,000 years. It has been determined for the EDU location based the executed analyses as the value of peak acceleration of the sub-surface of SL2= 0.06 g corresponding with a seismic event having 95% probability of not exceeding.

The design basis earthquake with the mean time between occurrences 1 x 100 years (determined from the respective calculations and the map of seismic risks of the Czech Republic) has been determined for the EDU location with the sub-surface acceleration of 0.050 g in a horizontal direction and 0.035 g in a vertical direction and corresponds to a seismic event with 90% probability of not-exceeding.

As the assignment for the EDU seismic upgrading in compliance with the international requirements, the value of the maximum design basis earthquake with the mean time between occurrence of 1 x 10,000 years at 0.1 g in a horizontal direction and 0.067 g in a vertical direction has been specified. At present, all units have been subject to seismic upgrade of all safety-relevant equipment and civil structures for the value of peak acceleration of sub-surface of 0.1 g.

2.1.1.2 Methodology used to evaluate the design basis earthquake

In the region of Central Europe and in the territory of the Czech Republic, there are no tectonic structures which might result in extremely severe earthquakes comparable to catastrophic earthquake in Japan that occurred on 11 March 2011. The territory of the Czech Republic represents a continental-plate area lacking any structures of the subduction zone type (subduction of the continental plates).

To determine the design basis parameters for the MVZ (SL2) level earthquake, three different approaches have been applied. The resulting values have been determined on the basis of comparison of all results generated by the applied methodological approaches as the most conservative values. Application of a combination of these methodological approaches should eliminate inaccuracies of the catalogues of earthquakes and the generalizations of the epicentre areas´ schemes and they should improve the reliability of the solutions´ results.

• Seismostatistic (probability) - prepared in two variants with the application of the same catalogue of earthquakes, however, different composition of the epicentre areas.

• Seismogeologic (seismotectonic) - based on the assumption that earthquake epicentres are related to active faults.

• Experimental - designated as the “zoneless method”, which does not require the definition of the source zones and delimitation thereof, nor determination of seismicity parameters and their seismic potential. It is based on measurement of actual characteristics of attenuation along the epicentre - the structure under review route.

The results of the SL2 determination by means of application of the individual methods:

1. Seismostatistic approach – method 1 (SL2 = 0.06 g)

2. Seismostatistic approach – method 2 (SL2 = 0.09 g)

3. Seismotectonic approach (SL2 = 0.06 g)

4. Zoneless method (SL2 = 0.05 g))

Seismistatistic approach - Method 1

Upon determination of the seismic risk, stability of tectonic as well as seismo-generating processes is assumed, which means that it is assumed that the so-far observed trend of seismic activity will remain unchanged in the future. In calculations, we always assume that earthquake can occur in any point of every area or active section of the fracture up to the maximum possible earthquake for such an area, such a fracture.

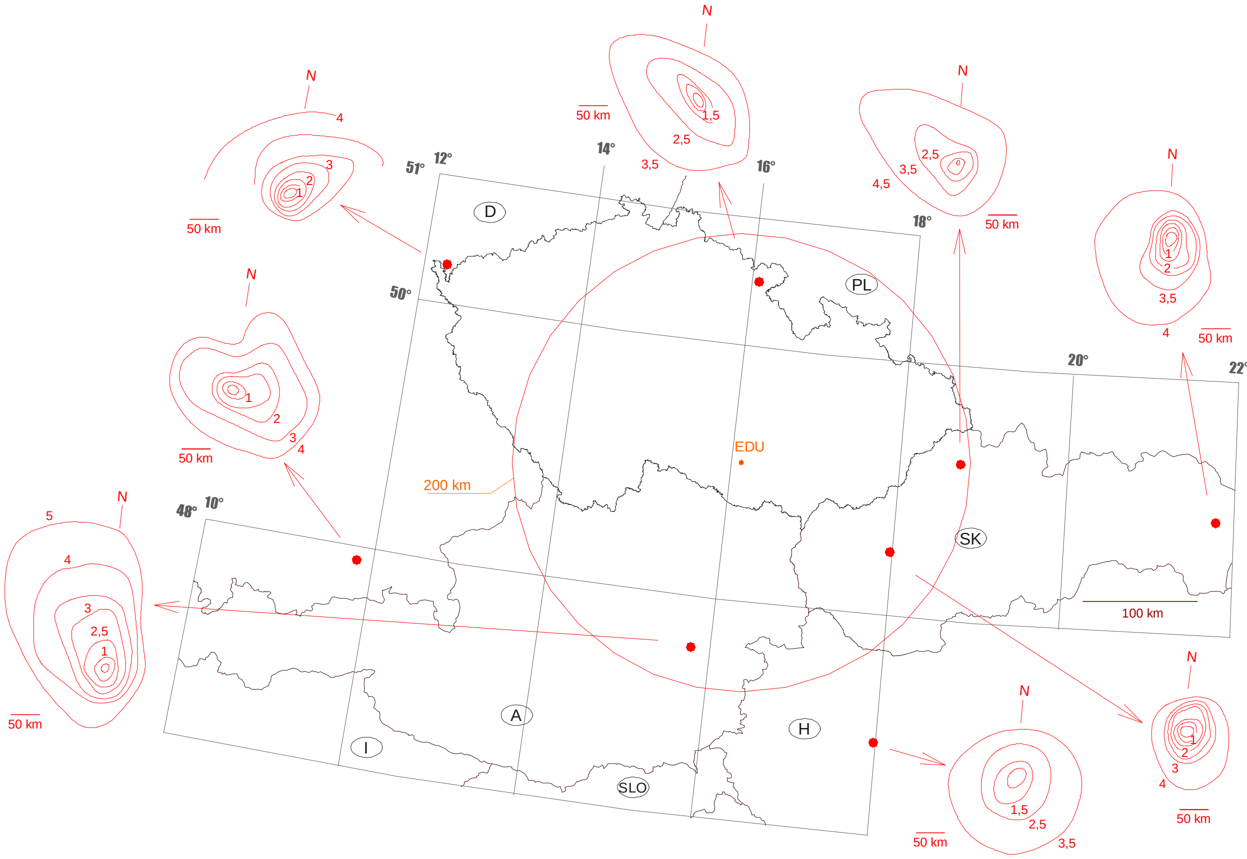
From the point of view of safety, it is therefore necessary to consider the least favourable case taking into account, on the one hand, the intensity of the maximum possible earthquakes in the individual epicentral areas and on the other hand, the shortest epicentral distance between the boundary of the epicentral areas or the active section of the fracture and the location.

For determination of seismic risk of the location, the two following approaches were applied based on the original IAEA 50-SG-S1 guideline and in accordance with the IAEA NS-G-3.3 recommendation:

- For the purpose of comparison, an expert appraisal based on the map of seismic zoning. The map´s application to nuclear facilities fails to fulfil the imposed requirements in a sufficient manner and therefore it is necessary to perform a more detailed survey.

- The probabilistic appraisal based on a theoretical mathematical model.

  - Fig. 2.1.1.2-1 Shapes of the macroseismic fields of the source areas in the region of the Dukovany NPP



Data concerning the maximum design value of macroseismic intensity for the location in view depending on the epicentral area for the Dukovany NPP´s location are determined by considering the epicentral areas, values of the maximum possible earthquakes, which the areas can produce in the time horizon of 10,000 years, and the attenuation curves of macroseismic intensities, which were derived from the epicentral area - location azimuth taking into account the shortest distances of the epicentral areas from the location (in accordance with the methodology currently applied in the nuclear industry, this concerns the most conservative appraisal).

<u>Seismistatistic approach - Method 2</u>

The approach applying Method 2 is based on calculation of seismic exposure using the probabilistic analysis of seismostatistic and partly seismotectonic input information (the probability curves of seismic exposure). This method enables to appraise the probability of the annual occurrence of vibrations having various intensity for many years ahead as well as to appraise the uncertainty such values are subject to.

The forecast of seismic events is based on the following source materials:

- Distribution of the source zones in the location and in the region.
- Seismicity of the source zones and maximum possible earthquake which may occur therein (seismic potential).
- Decrease of intensity of seismic movements of the ground with a distance from the epicentre to the location.
- Definition of the epicentral areas and seismicity thereof.

The prerequisite concerning the validity of the historical seismicity parameters even for future earthquakes is based on the idea of repeated rough drifts on the existing ruptures. However, experience says that new epicentres also occur in locations where no historical seismicity is recorded. This prerequisite forms one of the uncertainties concerning the input data.

The source areas of seismic exposure are, on the one hand, epicentral areas of historical earthquakes and on the other hand, lineaments of tectonic ruptures or crossings thereof. In Central Europe, to which the region where the Dukovany NPP is located belongs, 60 epicentral areas have been defined. Seismicity of these areas is expressed by the frequency diagrams and by values of the maximum possible earthquake thereof (seismic potential).

The other source zones are ruptures in the central part of the Český masiv (Czech massif), characterized by values of seismicity parameters appraised in an expert manner. The review of the ruptures together with seismic areas represents 71 epicentral areas in total. The seismic exposure of the Dukovany NPP has been executed based on the probable occurrence of earthquake in these source areas.

Seismogeologic (seismotectonic) approach

For the review of seismic activity of ruptures in the location´s area in view, the ruptures are divided, for the sake of clarity, into three classes and 6 categories with respect to the value of magnitude (Mmax) which they are potentially able to generate. The potential of the ruptures is assessed for the individual structural blocks - the regional geological units in a differentiated manner. The system of classification of ruptures into classes and the numerical coding thereof is based on the following overviews:

Class	Word designation	Category	M_{max}	I_0 [°MSK-64]
A	important seismogenic line	I	6.5	9.5
		II	6.0-6.4	9

B	important seismotectonic line	III	5.3-5.9	8
		IV	4.7-5.2	7
C	seismotectonic line	V	4.1-4.6	6
		VI	3.6-4.0	5

The data (maximum design values of macroseismic intensity I fro the Dukovany NPP´s location depending on seismoactive section of the rupture) are gained by considering the map of seismoactive ruptures, values of the maximum possible intensities of earthquake, which the seismoactive sections of the ruptures may generate in the time horizon of 10,000 years and the attenuation curves of macroseismic intensities, which were executed for the Dukovany NPP´s location taking into account the shortest distances of the sesimoactive sections of the ruptures from the location (in accordance with the currently applied methodology, this concerns the most conservative appraisal).

Experimental approach

The experimental determination of seismic exposure is based on the use of the so-called “zoneless method”. This method has a number of advantages and 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 has become applicable no sooner than lately when records acquired by instrumentation able to measure acceleration of seismic vibrations in the EDU´s location induced by earthquakes within regional distances are available. The new method thus is not based on subjective macroseismic data only (catalogues of historical earthquakes and the izoseist maps). If authentic instrumentation data are used, the uncertainties associated with the previously used methods arising from various empirical conversion formulas and expert appraisal are eliminated. For example, the relationship between the acceleration of seismic vibrations and the local macroseismic intensity is subject to uncertainties amounting up to two orders. Experts regard the new experimental method as reliable and perspective.

Combination of the aforementioned methods allowed eliminating inaccuracies thereof. As it follows from the results acquired by different methods, the realistic value of the maximum design basis earthquake of the SL-2 level for the Dukovany NPP´s location, which should not be exceeded within the time interval of 10,000 years with probability ≥ 0.95, is SL2=0.06 g.

The level of the design basis earthquake SL-1 has been determined for the Dukovany NPP based on calculations and the map of seismic exposure of the territory of the Czech Republic in correlation with the source materials for the preparation of the construction standard on the

Eurocode 8 basis. The maps of macroseismic intensities as well as the PGAH values show seismic exposure in the territory of the Czech Republic with 90% probability of not exceeding the intensity within the time horizon of 105 years for a period of observation 1,000 years.

Fig. Map of seismic exposure of the territory of the Czech Republic in values of macroseismic intensity

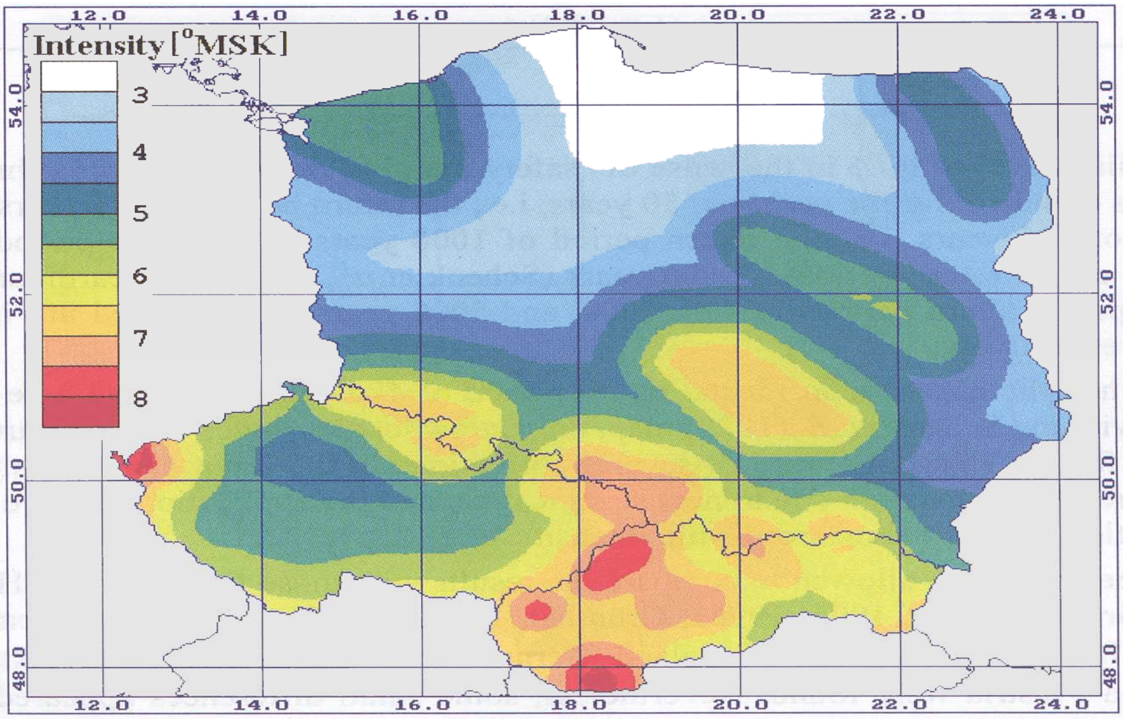
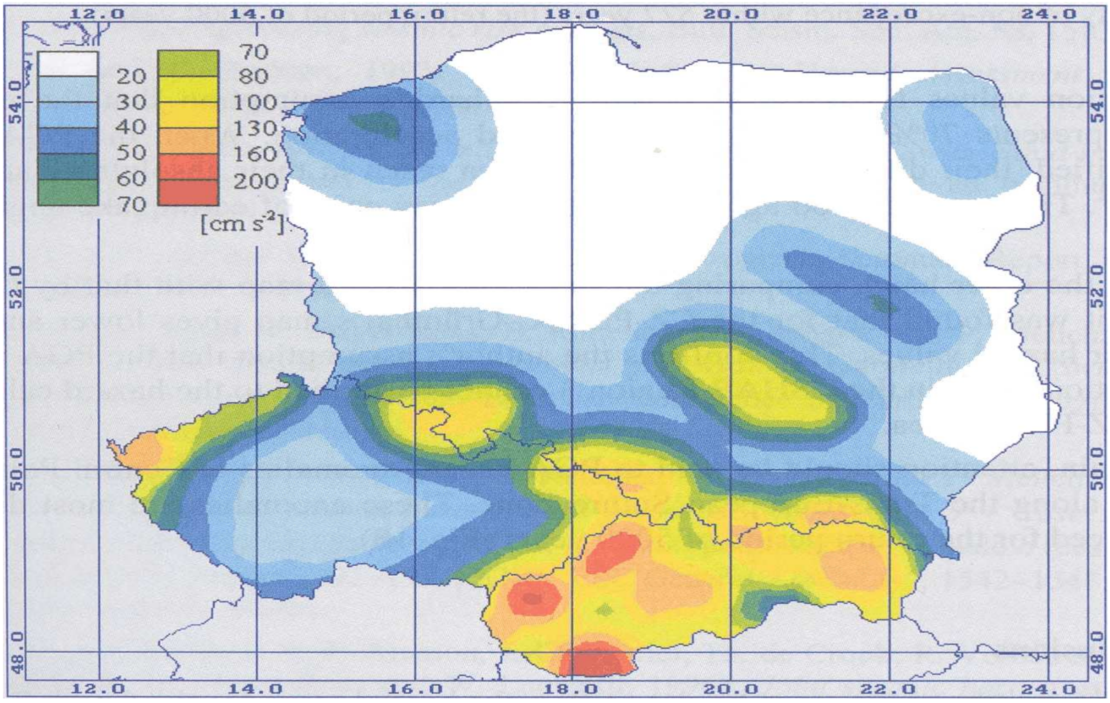


Fig.: Map of seismic exposure of the territory of the Czech Republic in values of PGAH



Based on the aforementioned maps and calculations, the SL-1 level of design basis earthquake for the Dukovany NPP has been determined at the level of intensity 6° of the MSK-64 macroseismic intensity scale and acceleration **SL1 = 0,05 g.**

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

The level of seismic risk to the location is determined by the value of maximum design basis earthquake with the time between occurrences 1 x 10 000 years. The actual values of seismic risk are 0.06 g (with 95% probability of not-exceeding within a time interval of 10,000 years), or 0.05 g (with 90% probability of not-exceeding within a time interval of 105 years) for a period of observation of 1,000 years. However, in compliance with the international requirements, the value of the maximum design basis earthquake with the mean time between occurrence of 1 x 10,000 years at 0.1 g in a horizontal direction and 0.067 g in a vertical direction has been determined.

The value of 0.1 g is the so-called "reference earthquake" and provides for the value on the seismic event intensity in respect of which the unit concerned should be upgraded. However, in reality, the occurrence of a seismic event the intensity of which is 0.1 g in the EDU location is not possible based on seismologic analyses and geological survey. The aforementioned facts can be substantiated by approximation of the curves characterizing the seismic risk to the EDU location towards values of higher intensities where the frequency of occurrence of a seismic event with intensity of 0.1 g can be estimated as lower than 1 x 10^{-8} event per year. This means one occurrence every 100 million years or more.

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 deterministic selection of equipment complying with the requirement for seismic resistance was based on the requirement concerning fulfillment of the selected safety functions: scram, provision of sub-criticality, residual output removal, after-cooling, I.O and II.O integrity, provision of sufficient I.O coolant reserve and provision of the containment leak proofness. To secure all the aforementioned functions, the necessary systems and equipment stated in the List of equipment for the EDU qualification have been selected. The listed equipment and civil structures in which the safety-relevant equipment is located were classified as belonging to "S" category based on the seismic resistance requirement. Depending on the required function, the "S" category equipment can be divided into the following sub-categories:

Sa (new labelling 1a) – Preservation of full functional competence in the course of and after earthquake up to and inclusive of the MVZ level is required.

Sb (new labelling 1b) – Preservation of mechanical strength and leak proofness after earthquake up to and inclusive of the MVZ level is required.

Sc (new labelling 1c) – Only seismic resistance from the point of view of possible seismic interactions and in particular preservation of position stability in the course of after earthquake up to and inclusive of the MVZ level is required

In accordance with the Methodology for the EDU seismic resistance assessment, assessment of seismic resistance of the civil structures and equipment belonging to “S” category has been executed by means of type tests, calculations or indirect assessment based on operational experience.

Table 1: The EDU´s civil structures classified in the S category subject to requirement of seismic resistance:

Designation	Name	Seismic category
800/1-01,02	HVB reactor building	Sb
805/1-01,02	Longitudinal intermediate room	Sb
806/1-01 through 04	Cross intermediate room	Sb
490/1-01,02	Turbine building	Sb
530/1-01,02	DGS	Sb
460/1-01,02	Vent discharge stack	Sc
593/1-01,02	SHN	Sb
1-4A2.3	Annex building electro - 4th system	Sb
581/1-01,02,05,06	Cooling towers	Sb
584/1-01,02	CČS	Sb
350/1-01,02 401/1-01,02	Electro - system ducts pipe ducts	Sb

Table 2: List of classified machine systems

Designation	Name	Seismic category
TC	Coolant purification PO-ŠOV-1	Sa
TF20,40,60	Intermediate circuit of SAOZ	Sa
TG	Storage pool cooling	Sa
TH10	Hydro accumulator	Sa
TH20,40,60	NT core emergency cooling	Sa
TJ20,40,60	VT emergency feed water system of PO	Sa
TQ20,40,60	Spray system of HZ	Sa
VF20,40,60	Essential service water	Sa
XL	Bubbler system	Sa

Designation	Name	Seismic category
YA	Primary circuit	Sa
YD	HCČ	Sa
YP	Pressurization	Sa
Demi 1MPa	Demineralised water 1 MPa	Sb
DGS 10,20,30	Dieselgenerator station	Sa
RHR system	RHR system for SO	Sb
Main steam	Main steam system	Sa
RČA	Quick-acting valves on the boundary of HZ	Sa
SHN	Emergency feed water system	Sa
TÚV	Thermal water treatment	Sb
TVD pump	TVD pumps for CČS	Sa
TVD SO	TVD distribution systems for SO	Sa
Vent. SO	Ventilation of control centres	Sa
HZ	Equipment on the boundary of the containment	Sa

Table 3: List of the classified electro systems

Designation	Name	Seismic category
1,2,3 DG	1st, 2nd, 3rd system of DGS 10, 20, 30	Sa
1,2,3 ZN II	1st, 2nd, 3rd safe (uninterruptible) power supply of category II, including the mode automatics	Sa
4 ZN II	4th safe (uninterruptible) power supply of category II	Sa
1,2,3 ZN I	1st, 2nd, 3rd safe (uninterruptible) power supply of category II	Sa
4 ZN I	4th safe (uninterruptible) power supply of category I	Sa
0ZN	Non-systems equipment (distr. DP10, DP20, DTE3, NDTE1)	Sa

Table 4: List of the classified SKŘ systems:

Designation	Name	Seismic category
BLOK DOZOR	Main control room	Sa
NOUZ DOZOR	Emergency control room	Sa
RTS	Reactor trip system	Sa
ESFAS	Engineered safety feature actuating system	Sa
EX - CORE	Measurement of neutron flux by means of external chambers	Sa
RLS	Reactor limitation system	Sa

Designation	Name	Seismic category
RCS	Reactor power control system	Sb
RRCS	Reactor rod control system (control system for safety and control rod assemblies)	Sa
SAS, SAS-N	Support interventions system	Sa
ELS	Emergency load sequencing	Sa
PAMS I, II	Post-accident monitoring system	Sa
ŘSBB	SKŘ at BD	Sa
ŘSBN	SKŘ at ND	Sa
ŘSBP	Relay automatics of the primary circuit	Sb
ŘSBT	Unit protection system	Sa
ŘSBS	Relay automatics of the secondary circuit	Sa

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

The design-determined value of SL2 is 0.06 g, while the actual upgrade of SSC is being performed to achieve the value of PGA = 0.1 g, which is recommended by the IAEA guidelines. The upgrade of the EDU´s design at a higher value that the actually determined one was decided on in 1995 in connection with the safety findings of the IAEA mission and by means of application of the then valid safety guideline IAEA 50-SG-S1 (1991). After full completion of the EDU´s seismic upgrade, the main safety margin will be given by the difference between the actual values of seismic exposure to the location and the design basis assignment of the seismic upgrade (PGA for the location 0.06 g compared to PGA_{SL2} = 0.1 g). Another safety margin that must be taken into account is the fact that these values are upper limits of the estimates for 95% probability of not-exceeding. Based on the executed analyses, the value of the maximum design basis earthquake SL2 was determined as 0.06 g while the actual upgrade of SSC is being performed to achieve the value of PGA = 0.1 g, which is recommended by the IAEA guidelines. The upgrade of the EDU´s design at a higher value was subject to a decision.

The walls of SFP belong to the seismic resistant hermetic zone and not even earthquake corresponding to the PGA=0.1 g design basis would result in the loss of integrity. The assessment has been executed based on reduced seismic effects due to the ductility coefficient, which means that the value of 0.1 g is the limit value as well.

The seismic assessment of the civil structures on the hermetic zone boundary has proven that even after a minor or medium LOCA accident in combination with other prescribed loads, leak proofness will not be aggravated even in case of occurrence of earthquake within 0.1 g.

After completion of the seismic upgrade, it is possible to consider the preservation of the NPP basic safety functions up to the sub-surface acceleration of 0.1 g in a horizontal plane and 0.067 g in a vertical direction.

2.1.2.3 Main operating provisions to achieve safe shutdown state

With all types of earthquake that come into question in the EDU location, provision of the following basic safety functions is not affected:

a) Reactivity control

b) Heat removal from nuclear fuel

c) Capture of ionizing radiation and radionuclides

The reactor would be scrammed by the RTS (HO-1) protection system automatically due to failures of the respective equipment, or manually by the button. Failure to scram the reactor due to mechanical defect due to a fall of control rods is covered by the so-called ATWS scenario, which has been analysed for EDU. After reduction of output by means of the reactivity feedback effects, the reactor would be shut down by injection of boron through at least one of the three safety injection systems.

After-cooling of the unit following a seismic event would be executed in the Feed & Bleed mode on II.O (feed water for SG through EFWP + steam removal through steam dump to atmosphere). The water reserve intended for after-cooling based on Feed & Bleed on II.O is given by the quantity of water in the demineralised water tanks (which have been seismically upgraded to 0.1 g as well). The water reserve in these tanks in case of heat removal through SG to the atmosphere should be sufficient for approximately 4 days. The reactors would be put to cold state and the method of long-term cooling would be subject to a decision following the respective inspection to determine damage caused to the equipment and civil structures (no regulation, procedure or manual with respect to the aforementioned has been prepared yet).

As an alternative, feeding SG using the FB means is considered, however, in this case problems concerning availability and transport thereof to the site concerned might occur. If, in the current state of seismically non-upgraded civil part of HVB, this method of after-cooling could not be applied, then it is possible to use the emergency method of after-cooling, the so-called Feed & Bleed on I.O (PRZR SV + ECCS pumps with the ESW support).

Heat removal from the storage pools would be provided in the same manner as prior to the event, i.e. by means of the TG system (100% redundancy).

The integrity of the containment is not endangered either - see the previous chap. 2.1.2.2.

2.1.2.4 Protection against indirect effects of the earthquake

The large tanks, which, in case of their destruction due to a seismic event, might flood out the civil structures containing safety-relevant equipment (e.g. feed water tanks and ESW tanks on II.O), have been assessed from the point of view of seismic resistance and have been anchored into the civil structures concerned sufficiently by means of the implemented modification in a manner eliminating the loss of integrity or undesirable interactions.

A seismic event, the intensity of which would be > 6° MSK-64 (PGAupper > 0.05g), would affect the seismic non-resistant equipment and civil structures, which would probably result in the full loss of power supply for own consumption of all the 4 units (the loss of the off-site power supply - 400 kV as well as 110 kV, unsuccessful switchover to the unit own consumption power supply by generators) with the subsequent scram of all 4 reactors and switchover to natural circulation. Due to the damage, NPP would likely lose the Jihlava pumping station and the raw water supply to the gravity-based water tanks.

In case of extensive damage to the infrastructure and long-term unavailability of the location (destruction of buildings, damage to communications, etc.), the altering staff might not be able to get to the location. In this case, the required activities would have to be performed by the staff present at the time of occurrence of the event. Altering would be resolved on an operational basis in cooperation with the state administration bodies (IZS, the army, etc.).

The communication means inside as well as outside the EDU´s premises will be endangered in a similar manner. Activities would be complicated in particular by loss of operability of the technical means of communication between the control centres and the persons performing the intervention, including communication with the external control centres and the authorities due to damage to the infrastructure surrounding the NPP.

Another difficulty will be availability of information concerning radiation situation within and on the boundary of the EDU premises. All the currently operated systems of radiation monitoring (CISRK, SEJVAL, SYRAD, TDS) are not in the seismic-resistant version, or parts thereof are located in PB civil structures with insufficient seismic resistance, which means that in case of earthquake > 6° MSK-64 (PGAhor > 0.05 g) their operation cannot be guaranteed. The radiation monitoring staff in civil structures having insufficient seismic resistance might be endangered as well and might not be available for measurement of radiation in an alternative manner using measuring devices.

Within the framework of the seismic analysis preparation in 2011, the risk contribution due to indirect effects of earthquake will be analysed by means of deterministic and probability assessment.

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 original design basis values for earthquake in the EDU location were changed in 1995 in relation to the safety-relevant findings of the IAEA mission. The new seismic assignment required reassessment of the safety-relevant equipment and civil structures onto a higher level of the reference earthquake PGA = 0.1 g and to implement seismic upgrade of non-compliant equipment and civil structures.

To maintain a permanent conformity of the current state of the equipment with the requirements of the design, a number of regular activities are being executed.

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

In the location, there are mobile means for fire suppression, which allow suppressing any type of fire. However, the means are deposited in the civil structure of the Fire Station, which has no seismic qualification. In case of damage to the Fire Station, the FB mobile equipment designed to suppress fires and to execute other necessary interventions might be unavailable. With respect to the fact that an occurrence of a seismic event is not fortuitous, however, that prior to such occurrence, it is possible to monitor some seismicity and symptoms, there will be enough time to move the equipment onto open areas where no damage due to destruction of civil structures or equipment is imminent. No procedures have been executed for such activity yet.

Access for the purpose of interventions by the brigades of the Fire Rescue Service (FB EDU, or external HZS) might be limited due to a fall of civil structures lacking the respective seismic upgrade onto the internal access communication as well as due to a fall of debris onto the entrance area of the power plant. As an alternative, it could be possible to use the service entrance to the plant premises.

In the location, 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. Availability of the main generating units for mobile equipment was solved in cooperation with the authorities (IZS, Armed Forces of the ČR, etc.).

It is likely that it would not be possible to use the emergency preparedness shelters or the Emergency Commission workplaces (EC), or the workplaces of the Technical Support Centre (TSC), which are located under civil structures with insufficient seismic resistance. The inaccessibility of the TSC could be partly solved by its operation from the emergency control room (so far there are no detailed instructions available).

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

Based on the extraordinary inspections concerning seismic resistance, which were executed after the accident in the Fukushima nuclear power plant with respect to seismicity, no major non-conformities of the current state with the design basis requirements have been discovered. The upgrade of SSC is being performed to achieve the value of 0.1 g, which is a higher value than the maximum design basis earthquake = 0.06 g.

2.2 Evaluation of safety margins

2.2.1 Range of earthquake leading to severe fuel damage

After the seismic upgrade of the civil structures, the limit for the loss of basic safety functions will increase up to the level of earthquake the intensity of which > 7° MSK-64 (PGAupper > 0.1 g). A seismic event of intensity > 7° MSK-64 (P GAupper > 0.1 g) might result in the loss of the NPP safety functions. Nevertheless, this does not concern a value, which has a character of the “cliff edge” boundary, as each item of equipment has a different value of boundary seismic resistance with different margin against 0.1 g. With respect to the safety margins, it is assumed that some of the safety systems would remain in the state able to execute the respective safety functions.

After-cooling in case of a seismic event > 0.1 g would continue to be realized by means of the Feed & Bleed method on II.O (the SG water feeding through EFWP + steam removal through steam dump to atmosphere), or by emergency after-cooling using Feed & Bleed on I.O (PRZR SV + ECCS with the ESW support). All the aforementioned systems and components have been seismically upgraded to 0.1 g and have minimum 100% redundancy (EFWP) and are power supplied from the electrical emergency power system (from the seismically upgraded DGs).

The deterministic limit value of the seismic event intensity, upon exceeding of which damage to core can be expected, it is possible to take the value of 0.112 g, which is the limiting resistance of CHV.

2.2.2 Range of earthquake leading to loss of containment integrity

The seismic assessment of the civil structures on the hermetic zone boundary has proven that even after a minor or medium LOCA accident in combination with other prescribed loads, leakproofness will not be aggravated even in case of occurrence of earthquake within 0.1 g.

2.2.3 Earthquake exceeding DBE and consequent flooding

The EDU location is not subject to the risk of flooding due to natural or special floods. The plant premises are located on an elevated plane at the altitude of 383.5 - 389.10 m above sea level and the main civil structures where the safety-relevant equipment is located are located at the altitude of 389.10 m above sea level. The roof-like water drain directs at the deep-sunken watercourses of the Jihlava and the Rokytná rivers running offwards the power plant.

Dalešice reservoir is located upstream the Jihlava river - the dam of the reservoir is within a distance of 4 km upstream the river from the power plant. The dam crest is at the altitude of 384.00 m above sea level and maximum level of water in the reservoir during throughflow of high waters in the reservoir is at the altitude of 381.50 m above sea level. The hypothetical break wave from Dalešice reservoir does not endanger the NPP premises as such with respect to its altitude which is approx. 8 m higher than the maximum level of the reservoir.

Mohelno reservoir, the course of the profile of which is nearest to NPP at a distance of approx. 1 km in the valley under the power plant, has the dam at a distance of approx. 2 km up the river. The altitude of the dam crest is 307.15 m above sea level and maximum water level is 303.30 m above sea level, which is by approx. 80 m lower than the EDU dimension figure of +0.0.

In case of loss of the Jihlava pumping station due to a break wave resulting from damage to the dam during a hypothetical earthquake, the EDU´s location is equipped with sufficient reserves of water to perform residual heat removal - see chapter 5.2.2 for further details.

The other surface watercourses in the surroundings of NPP are streams and small rivers the throughflows of which are lower by two orders compared to the Jihlava river. This means that in the surroundings of EDU there are no water resources that might cause flooding of the NPP´s location in case of earthquake.

2.2.4 Measures which can be envisaged to increase robustness of the plant 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 EDU shall not exceed, with a 95 % probability, 6°MSK-64 (PGAupper = 0.06 g). The actu al resistance of buildings, systems and structures is higher, thus safety margin is available for the remaining 5 % uncertainty.

In stipe of this fact, as early as in 1995, it was decided to perform upgrade of the EDU´s safety-relevant equipment and civil structures to achieve the value of peak acceleration of the sub-surface PGA = 0.1 g (the maximum design basis earthquake, MDE/SL2/SSE). More than 90 % (among others, all technological equipment) of the safety-relevant equipment (based on the qualification list) has already had the compliant qualification documentation proving seismic resistance, and for the remaining equipment (the electro part and I&C), the works relating to the implementation of modifications are being finalized.

The aforementioned hypothetical consequences of earthquake will be eliminated by strengthening the in-depth defence level during earthquake, the consequence of which may be the loss of ability to perform safety functions by the implementation of measures aiming at securing non-technological auxiliary functions (the civil structures´ accessibility, fire-suppression equipment availability, provision of shelters)..

Activities would be complicated in particular by loss of operability of the technical means of communication between the control centres and the persons performing the intervention, including communication with the external control centres and the authorities due to damage to the infrastructure surrounding the NPP.

The purpose of the proposed provisions is further upgrade of the in-depth defence levels during earthquake.
The opportunities for improvement of the in-depth defence are provided in the following table. The table contains also areas for which it is necessary to complete supplementary analyses as at the time when the review was executed they were unavailable. A more detailed description of the corrective actions is provided in the annex.

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
Seismic resistance SKK	Complete the design of the EDU´s seismic upgrade	II	Under implementation
Seismic resistance SKK	Inspection and provision of anchoring for non-seismic equipment	I	
Regulations	Execute operating procedures for earthquake	I	
Regulations	EDMG guidelines for the use of alternative means	II	
Emergency preparedness	Provide for the functioning of the emergency response teams in case of HŘS´s unavailability	I	
Analyses	Resistance of the HZSp building with respect to seismicity.	I	Under implementation
Communications	Alternative means of communication after a seismic event	I	
Staff	Analysis of exposure of the shelters during a seismic event	II	
Staff	Provision of sufficient staff after a seismic event	I	
Analyses, technical means	Access to civil structures, availability of heavy equipment	II	
Analyses	Seismic PSA		PSR finding, under implementation

3 Flooding

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 EDU location is not subject to the risk of flooding due to natural or special floods. The plant premises are located on an elevated plane at the altitude of 383.5 - 389.10 m above sea level and the main civil structures where the safety-relevant equipment is located are located at the altitude of 389.10 m above sea level. The roof-like water drain directs at the deep-sunken watercourses of the Jihlava and the Rokytná rivers running offwards the power plant. The local superelevation of the surface in the area of western towers is resolved by draining possible waters from rural areas by means of the circumferential drainage ditch into the storm-water drainage system.

The Jihlava river is nearest watercourse and it is used as a source of technological make-up water for the power plant. The Jihlava river, including the system of Dalešice - Mohelno reservoirs, runs to the north of the power plant, from the northwest to the southeast and the shortest distance between the power plant and the river is approximately 1 km. Extraction of technological water is executed from Mohelno reservoir, which serves as the balancing reservoir for Dalešice reservoir as well.

Dalešice reservoir is located upstream the Jihlava river - the dam of the reservoir is within a distance of 4 km upstream the river from the power plant. The dam crest is at the altitude of 384.00 m above sea level and maximum level of water in the reservoir during throughflow of high waters in the reservoir is at the altitude of 381.50 m above sea level. The dam´s height is 88 m above the bottom.

Mohelno reservoir, the course of the profile of which is nearest to NPP at a distance of approx. 1 km in the valley under the power plant, has the dam at a distance of approx. 2 km up the river. The altitude of the dam crest is 307.15 m above sea level and maximum water level is 303.30 m above sea level, which is by approx. 80 m lower than the EDU+0.0 dimension figure.

The other surface watercourses in the surroundings of NPP are streams and small rivers the throughflows of which are lower by two orders compared to the Jihlava river. Small ponds are built on the aforementioned watercourses, the largest of which have surface areas of approx. 0.5 ha and they are located in a lower altitude than the power plant premises and thus represent no risk to the premises concerned.

The hypothetical break wave from Dalešice reservoir does not endanger the NPP premises as such with respect to its altitude which is approx. 8 m higher than the maximum level of the reservoir. Of the power plant equipment, which is endangered by a potential possibility of flooding due to floods in of throughflow of high waters in the Jihlava river, only the raw water pumping station on the Jihlava river can be affected. The raw water pumping station provides for supply of industrial make-up water for the EDU operation.

Up to now, the technical and safety inspections of the Dalešice-Mohelno water reservoirs system, no significant findings or facts have been discovered that would imply any danger to the water reservoirs´ safety.

The design basis values of maximum daily precipitation amounts by frequency of reoccurrence are as follows:

Reoccurrence interval [number of years]	100	1000	10 000
Daily precipitation amount [mm]	77	93	115

The underground water level in the EDU premises is several meters under the foundations of the civil structures. Refilling and origination of reserves of underground waters in the territory concerned occurs exclusively due to the atmospheric precipitation infiltration. Natural drainage occurs to the north and to the south towards the Jihlava and the Rokytná watercourses. The local superelevation of the average level of the underground water level in some of the civil structures is resolved by pumping from drill holes to the waste-water disposal system. Endangering of the civil structures or rooms where the nuclear-safety-relevant equipment is located due to shallow horizon of underground waters is excluded.

3.1.1.2 Methodology used to evaluate the design basis flood.

The EDU location territory, according to the Quitt´s classification of climatic regionalization, belongs to slightly warm climatic regions. The annual course of precipitation in in the long-term characterized by the highest precipitation amounts in summer months with the maximum in June (70 mm) and the lowest precipitation amounts in winter months with the minimum in January (21 mm).

The waste-water disposal system is designed as a branching network providing for removal of atmospheric water in a gravity-induced manner from the area of approx. 80 ha, which is connected to the main atmospheric-water main sewer in front of the EDU premises.

The catchment areas of the individual storm sewers were based on the hydro-technical situation and these catchment areas were assigned the respective flow rate and run-off coefficient depending on the method of development and type of land. The respective profile of the storm sewer was designed subsequently with respect to the design basis quantity. Based on hydrotechnical calculations concerning the storm-water disposal system, the amount of atmospheric water has been determined for periodicity p = 1-minute and 15-minute equivalent rain with intensity of i = 135 l/s.ha and amounts to Qrain = 3,028 l/s, it being understood that the storm-water disposal system´s capacity is 3,810 l/s.

Load due to climatic incidents (in general) is based on the statistical processing of annual extreme values of the relevant meteorological quantities measured in the course of at least 30 years in the EDU location and in the meteorological stations in the surrounding region, which have the same character as the EDU location as far as climatic conditions are concerned. The methods of statistical processing are based on the document issued by the IAEA titled “Safety Standards Series” [Safety Guide NS-G-3.4: Meteorological events in site evaluation for Nuclear power Plants], IAEA 2003 based on the application of the Gumbel distribution.

During the scheduled service life of EDU, it is not realistic to consider any changes in the location concerned which might affect the operational safety.

The assessment of effect of torrential rains on the NPP safety is based on very conservative premises that all the storm-water drains will be blocked (with the exception of downcomers on civil structures) and that one-day precipitation with repeatability of once every 10,000 years when the precipitation amounts to 115 mm/24 hours will occur. The assessment of the rain effect is based on the cumulative collection of the total precipitation amount in 24 hours (115 mm) on the ground space of the power plant and on the altitude survey of the ground space and the communications within the power plant premises.

The assessment of effect of torrential rains on the NPP safety is based on very conservative premises that all the storm-water drains will be blocked (with the exception of downcomers on civil structures) and that one-day precipitation with repeatability of once every 10,000 years when the precipitation amounts to 115 mm/24 hours will occur. The assessment of the rain effect is based on the cumulative collection of the total precipitation amount in 24 hours (115 mm) on the ground space of the power plant and on the altitude survey of the ground space and the communications within the power plant premises.

Of the civil structures where the safety-relevant equipment is located, the lowest located one is the Annex building of the emergency feed water system, which is by 14 cm higher than the surrounding ground space, followed by DGs I, which is by 17 cm higher than the surrounding ground space, which represents minimum reserve in case of hypothetical increase of water level on the surrounding ground space by +11.5 cm.

The operational experience concerning start-up and operational utilization of nuclear power stations with VVER reactors includes even a potential risk to the selected systems and equipment of NPP, essential for the NPP safety, due to internal flooding. The basic basis to assess consequences of internal inundation for the EDU operational safety are results of the works currently executed in the respective area, results of the EDU equipment qualification, experience and approaches to resolving the problem concerned in the VVER nuclear power stations of a similar type. As it follows from the analyses, these events do not prevent safe shutdown and after-cooling of the reactor units.

3.1.1.3 Conclusion on the adequacy of protection against external flooding

Flooding from the watercourses does not endanger the NPP premises as such due to its altitude and all the civil structures located within the EDU premises are safely protected by the altitude of the premises.

The potential possibility of flooding due to inundations resulting from high waters going down the Jihlava river is relevant only for the raw water pumping station on the Jihlava river (provides for supply of make-up water necessary for the EDU operation).

Flooding of the civil structures essential from the point of view of safety from the gravity-induced storm-water disposal system is not possible with respect to its regular maintenance. Even in case of occurrence of theoretically possible short-term precipitation with a higher intensity, the entire passive gravity-induced storm-water disposal system is able to drain such precipitation with respect to the large volume of sewers and the short period of duration of such intensive precipitation.

The EDU location position excludes the risk of flooding due to natural or special floods. The EDU civil structures are designed as resistant to flooding upon maximum one-day precipitation amount, at which on the surface in the altitude of 389.1 m above sea level, the maximum level of water staying is 115 mm, which is the total precipitation amount for 24 hours upon the 10,000-year maximum. In addition, the current geodetic surveys in the area of the entrance and the surrounding ground space in the vicinity of the civil structures essential from the point of view of safety, provide for a justifiable base of the statement asserting that even in case of absolute failure of the storm-water disposal system´s function, a continuous surface of water reaching the level of 115 mm could occur on the areas subject to review. In spite of the aforementioned, the EDU´s design is dimensioned for this level with a margin.

As it follows from the analyses, internal flooding does not prevent safe shutdown and after-cooling of the reactor units.

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

Of the equipment essential from the point of view of safe shutdown and after-cooling of the reactor units and solution of possible abnormal and failure conditions of the EDU reactor units, the following civil structures and SSC located therein could be endangered in case of flooding due to extreme precipitation:

1. Turbine halls of HVB I and HVB II (SO 490/1-01, 02) – emergency (auxiliary) feed water pumps.
2. Diesel-generator plants of HVB I and HVB II (SO 530/1-01, 02) – auxiliary circuits, generator exciting.
3. Central pumping station (SO 584/1-01, 02) – essential service water pumps and fire water pumps.
4. Cooling towers (SO 581/1-01 ÷ 08) - cooling service water overflowing.
5. Annex building of the emergency feed water system (SO 593/1-01, 02) - emergency feed water system pumps, 1 MPa demineralised water pumps and 0.4 MPa demineralised water pumps.

It was stated that all the aforementioned civil structures cannot be endangered due to floods. The only civil structure containing equipment which relates to the EDU operation and which might be affected by flooding resulting from the watercourses is the raw water pumping station situated on the right bank of Mohelno equalizing reservoir with the altitude of 303.80 m above sea level. Dalešice reservoir is designed for safe transfer of Q1000 ("once-in-a millennium flood"), which is for Mohelno 460 m^3/s. At this flow rate, the water level in the reservoir does not exceed 303.30 m above sea level. Therefore, there is still 50cm reserve before problems concerning flooding in the respective civil structure occur.

In case of failure of the dam of the upper accumulation reservoir it is necessary to expect flooding of the raw water pumping station and the loss of raw water supply of the power plant.

The raw water pumping station is not classified among the safety systems and the loss of the function concerned is resolved in the operating procedure as abnormal situation - requirement to shut down all reactor units. Time, for which the current reserves of raw water in the power plant will be available before reaching the minimum level for the ESW pumps work, which means preservation of the UHS function (Cliff Edge Effect) is minimum 400 hours.

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

The basic design basis provisions to prevent occurrence of floods due to precipitation, in addition to the power station localization, is the sufficiently dimensioned storm-water disposal system, the altitude disposition of entrances, accesses and gates with respect to the surrounding ground space and weathering of the adjacent communications and other outdoor areas adjacent to civil structures essential from the point of view of nuclear safety. All the civil structures located in the EDU premises are safely protected by means of the altitude level of the premises.

3.1.2.3 Main operating provisions to prevent flood impact to the plant.

Upon external floods, performance of not a single one of the following basic safety functions is endangered:

a) Reactivity control.

b) Heat removal from the nuclear fuel.

c) Capture of ionizing radiation and radionuclides.

The core reactivity controls independent of external floods and provides for sufficient sub-criticality in case of possible floods.

In case of external flood (torrential rain), the function of heat removal is secured through the system of residual heat removal from core pursuant to procedures for abnormal or accident situations (standard after-cooling, emergency one using Feed & Bleed on the secondary or primary side).

Residual heat removal from spent nuclear fuel deposited in the pools designed for storage thereof SFP is dependent, on the one hand, on the function of the equipment located in the reactor buildings and on the other hand, on the function of the essential service water (ESW) system, including the cooling towers - the so-called “ultimate heat sink” (UHS). In case of external flood, these systems are not affected.

The last barrier against radioactivity leakage from core, i.e. the containment integrity, including the vacuum bubbler system, cannot be endangered due to flood. The insulation of its piping as well as penetrations is provided by means of redundant separating components, which cannot fail due to flood either.

Contrary to floods due to external causes, internal floods have usually only local character or can be managed easily (by deactivation of pumps) and with redundant safety systems, they are regarded as one of the possible causes of the loss of the safety function concerned in the respective division; with non-essential systems, one space flooding means the loss of the respective technological function on the affected reactor unit. For this reason, internal floods are resolved as abnormal situations in the operating procedure.

In case of a long-term loss of the SFP cooling system due to internal flood, it is not possible to exclude leakage of radioactive media from the storage pools.

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

Occurrence of excessive precipitation in the Jihlava river catchment area is usually accompanied by its swelling directly in the central part of the city of Třebíč, which means difficulties concerning in particular transport from the left-bank districts of the city (where most members of the EDU staff and employees of the supply companies live) to the right-bank districts thereof (where communications to the plant are located). It is also necessary to take into account local restriction of passability of the communications in the neighbourhood of NPP due to mudslide from the surrounding fields.

In case of floods and long-term unavailability of the location, the altering staff might not be able to reach the location operationally. In that case, the required activities 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.).

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 FB wireless stations to the Fire Rescue Service in Třebíč. In this manner, the long-term communication between the persons executing interventions and the control centre might be endangered as well as the communication with the off-site centre of the administrative authorities (Crisis Centrum SÚJB, Regional Emergency Commission, IZS, 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

To secure protection against floods from external sources, a number of regular activities are being executed for the maintenance of the required condition of the equipment concerned. Periodic checks, maintenance of the storm-water disposal system and the schedule for cleaning cavities provide for its design basis parameters. Inspection of the technical condition of the sewer lines is executed once a year and necessary repairs are provided depending on the discovered condition. This concerns e.g. inspection of the grates and intercepting traps; possible repairs or replacement is executed according to discovered condition thereof.

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

In the EDU location, the FB is available, which disposes of the respective equipment and the members of which are trained to suppress any fire as well as to pump water at any point of the location. The fire-suppression equipment and the staff are located in the Fire Station civil structure, which is not anyhow protected against flooding, however, it is not assumed that in case of flood in the EDU location, use of the mobile fire-suppression equipment would be eliminated.

Independent means for transport and pumping media are the FB mobile equipment as well, which is modified to pump water in case of floods as well.

Access to certain civil structures might be limited due to floods. With the right function of the waste-water disposal system, there is no risk that the access might be blocked, nevertheless, the CČS I civil structure is located lower than the 0.0 m level and possible increased amount of water might accumulate near the pools under the cooling towers. These situations have not been analysed yet and no instructions concerning the way how to proceed are available.

It is likely that it would not be possible to use the emergency preparedness shelters or the workplaces of the EC and the TSC, which are not protected against flooding. Possible unavailability of the shelters would be resolved on an operational basis by evacuation of the staff out of the location.

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

During inspections and checks initiated by the accident that occurred in the Fukushima nuclear power plant, some minor deviations from the assumed condition of the equipment have been discovered and there are being removed in the work order.

3.2 Evaluation of safety margins

3.2.1 Estimation of safety margin against flooding

The EDU location position excludes the risk of flooding due to natural or special floods. Internal floods, due to their local character and easy management, do not endanger nuclear safety (abnormal situation in the operating procedure).

Although the geographical position of the EDU´s location excludes exposure due to flooding from natural or special floods, the EDU´s civil structures are designed as resistant to flooding during extreme rain precipitation if the waste water disposal system were out of order completely. Even the hypothetical case, when all rain precipitation would not flow off to the lower situated areas in the surroundings of EDU and would form continuous surface exceeding the maximum design basis value 115 mm, does not have a character of limiting conditions as the footing of most of the civil structures containing equipment important from the point of view of safety are situated even above the surrounding terrain (e.g. building of the emergency feed water system, which is by 14 cm higher than the surrounding ground space, followed by DGs I, which is by 17 cm higher than the surrounding ground space.) which represents further minimum reserve of approx. 20 % compared to compared to the total amount of rain precipitation for 24 hours at the 10,000-year maximum.

Consequences of floods might result in inability to manage consequences of floods concerning threat to persons, impossibility to use ECC and the emergency preparedness shelters and communication passabilities due to extensive floods and damage to the infrastructure in the EDU surrounding environment.

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

The purpose of the proposed provisions is further upgrade of the in-depth defence levels during floods. The opportunities for improvement of the in-depth defence are provided in the following table. The table contains also areas for which it is necessary to execute supplementary analyses as at the time when the review was executed they were unavailable. A more detailed description of the corrective actions is provided in the annex.

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
Regulations	EDMG guidelines for the use of alternative means	II	
Analyses	Analysis of exposure of the shelters during floods	II	

4 Extreme weather conditions

4.1 Design basis

4.1.1 Reassessment of weather conditions used as design basis

The original design basis arises from the PIN AE-5.6 Russian standard, according to which meteorological phenomena with the time of re-occurrence of 10,000 years must be taken into account. If sufficient meteorological source data concerning the respective location are not available and if the parameters are not determined for the time of re-occurrence of 10,000 years, the load for extreme wind should be increased by the factor equal to 2.5 and for extreme snow by factor 2.0. However, prior to the EDU´s construction no specific values of the design parameters were determined and ČSN for standard civil structures were taken as a basis.

In 2000, the design basis parameters of climatic incidents for the EDU location were reassessed so that the load due to climatic incidents (in general) was based on the statistical processing of annual extreme values of the relevant meteorological quantities measured in the course of at least 30 years in the EDU location and in the meteorological stations in the surrounding region, which have the same character as the EDU location as far as climatic conditions are concerned. For the design basis load due to climatic effects, repeatability of occurrence of the respective incident once every 100 years is considered. For extreme design basis load due to climatic incidents, repeatability of the respective incident once every 10,000 years is considered.

The methods of statistical processing are based on the document issued by the International Atomic Energy Agency titled "Safety Standards Series" Safety Guide NS-G-3.4: Meteorological events in site evaluation for Nuclear power Plants, IAEA 2003). Civil structures in seismic class "S" shall withstand the effects of an extreme design basis load so that the function of systems essential for nuclear safety is not put at risk. The other civil structures important from the point of view of safety are subject to definition of design basis climatic action. Determination of parameters for 100-year and 10,000-year climatic action is executed according to the IAEA NS-G-3.4 guideline with the use of the Gumbel distribution. The design basis as well as extreme load applicable to the civil structures are then determined from the values of meteorological parameters.

In 2009-2010, calculation check of the limit values of resistance of the safety-relevant civil structures to extreme wind and snow conditions was executed. The calculations of the limit values of the civil structures´ load included check of internal forces of the individual main load-bearing elements of the structure concerned for the most unfavourable combination of loads. In cases where the calculated values were lower than the design basis ones or the extreme load, effect on the equipment located in the civil structures concerned and on the safety functions performing by the respective equipment was subject to review as well.

For the evaluation of the resistance of the civil structures and the equipment to the effects of the other climatic incidents, the license documentation regards the following extreme climatic effects:

- Wind
- Snow/Ice
- High/Low temperature

The parameters for 100-year and 10,000-year load (design basis and extreme load) due to effects of climatic incidents are provided in the undermentioned table:

Event (climatic incident) / Parameter	**Time of re-occurrence 100 years**		**Time of re-occurrence 10,000 years**	
	Value	**Load**		**Value**
Gust wind / speed	46,2 m/s	0,69 kN/m^2	60,6 m/s	1,26 kN/m^2
Snow / recalculated water column	109,0 mm	1,09 kN/m^2	195,0 mm	1,95 kN/m^2
Maximum temperature/ abs. max / year 6-hour average	39,0 ℃ 38,5 ℃		46,2 ℃ 46,2 ℃	
Minimum temperature / abs. min / year Daily average 5-day average	-30,8 ℃ -24,0 ℃ -21,4 ℃		-46,7 ℃ -37,8 ℃ -35,3 ℃	

The value of the wind stated in the table corresponds to the value of non-repetitive (peak) speed of wind. This speed forms a basis of the calculation of the mean speed of wind for the integration interval of 10 s, for which the calculations of civil structures are executed. The mean value for the time of re-occurrence equal to 100 years is 27.4 m/s and for the time of reoccurrence of 10,000 years corresponds to the value of 49.1 m/s.

The review of resistance to extreme values of meteorological events was executed for the safety-relevant civil structures belonging to seismic category S.

Designation	Name	Seismic category
800/1-01,02	HVB reactor building	Sb
805/1-01,02	Longitudinal intermediate room	Sb
806/1-01 through 04	Cross intermediate room	Sb
490/1-01,02	Turbine building	Sb
530/1-01,02	Diesel-generator station (DGS)	Sb
460/1-01,02	Vent discharge stack	Sc

593/1-01,02	Annex building of SHN	Sb
1-4A2.3	Annex building electro - 4th system	Sb
581/1-01,02,05,06	Cooling towers (CHV)	Sb
584/1-01,02	Central pumping station (CČS)	Sb
	Demineralised water tanks 1000 m^3	Sb

4.1.1.1 High winds

The original assessment of the civil structures with respect to load due to wind was executed by means of the deterministic approach in 2000 and by means of the probabilistic approach in 2008. The unsatisfactory results raised suspicion that the design characteristics are determined in a too conservative manner and that the data concerning speed of wind acquired from the meteorological station in Kuchařovice are overestimated.

For the update of meteorological data, data from five stations in the EDU surroundings for a period of 50 years have been obtained. The review performed in 2010 revealed that the original data were not overestimated, on the contrary, the executed review showed that gust wind is somewhat higher and that the determined load of the civil structures due to wind is considerably higher.

Compared to the basic pressure of wind for the given location pursuant to ČSN 73 0035, which has been determined for standard civil structures at the value of 0.45 kN/m^2 the respective wind pressure in case of the design basis load is twice the size thereof and in case of extreme load even 3.8 times higher.

4.1.1.2 Heavy snowfall and ice

The input values for extreme values of snow were determined on the basis of data acquired from the nearby meteorological station Hrotovice.

Compared to the basic weight of snow for the respective location pursuant to ČSN 73 0035, which has been determined for standard constructions in locations to the west of Dukovany at the value of 0.7 kN/m^2, this value is, in case of design basis load, 1.56 times higher and in case of extreme value 2.78 times higher.

The considerably higher values of load due to snow compared to the normative load are given by the fact that the snow map in ČSN 73 0035 states the hundred-year value divided by a factor equal to 1.7 and the load due to snow with the time of re-occurrence is not taken into account for standard civil structures at all.

4.1.1.3 Maximum and minimum temperature

The input set of measured data for extreme load due to outdoor temperatures was selected from measurements of outdoor air temperatures in the meteorological stations in

Kuchařovice, Moravské Budějovice and Dukovany. For the purpose of evaluation, the annual maximum and minimum temperatures in the calendar years were considered.

4.1.1.4 Consideration of potential combination of weather conditions

For the solution of combined loading conditions, the calculations for individual loading conditions were executed first. Results of the solutions of the individual loading conditions were combined in loading combinations. The 13 basic loading conditions were determined - these conditions were always subject to service load, which means permanent, random long-term and partial short-term loads + load due to a single climatic event. Load due to climatic events with the time of re-occurrence of 100 years as well as with the time of re-occurrence of 10,000 years was taken into account. The design of combinations was based on the principle under which the operational load is always combined with one extreme load (which means with the one having the time of re-occurrence of 10,000 years). The combination with the load due to climatic events with the time of re-occurrence of 100 years was not executed as this is included in the prerequisite for extreme load. In these calculations, factors of load reliability were taken as equal to 1.0. With respect to the fact that a combination with extreme load is concerned, the factor of load reliability for these combinations is taken as equal to 1.0 (similarly, as in case of combinations with seismic load). However, at the same time, the factor of combination pursuant to ČSN 73 0035 is not taken into account and the combination of loads with extreme events is not decreased by this factor. In the calculations, inner forces of individual main load-bearing elements of the structures for the most disadvantageous combination of loads were checked.

The issue of the concurrent occurrence of two extreme climatic incidents in the EDU location was resolved in 2008 based on the qualitative and quantitative analysis of the climate dynamics in the Czech Republic. The analysis applied the following principle:

Concurrent occurrence of two extreme climatic incidents is conditioned by mutual dependencies of the two incidents. Such dependence in the occurrence of a specific meteorological situation in the EDU location enabling occurrence of the two or all the considered climatic incidents with concurrent occurrence. Based on the analysis, only the following two combinations of extreme climatic incidents were designated as relevant from the point of view of risk:

Combination of extreme incidents	**Upon climatic incident occurrence**	**Probability of occurrence**	**Risk**
a) high temperature b) high wind	SWa, Sa	2 % in summer months (3/12 of the year)	Similar consequences as individual incidents a) or b), however

Combination of extreme incidents	Upon climatic incident occurrence	Probability of occurrence	Risk
			with lower frequency of occurrence
a) high wind b) high precipitations (snow)	Ec, Wc, NWc	25 % in winter months (3/12 of the year)	Risk as in a). High wind does not allow formation of a high layer of snow on the roofs of the civil structures concerned.

Wc	Western cyclonic situation
NWc	North-western cyclonic situation
Ec	Eastern cyclonic situation
Sa	Southern anticyclonic situation
Swa	South-western anticyclonic situation

4.2 Evaluation of safety margins

4.2.1 Estimation of safety margin against extreme weather conditions

With respect to realistic values of climatic conditions occurring in the EDU location and real resistance of safety-relevant civil structures to extreme conditions, performance of all basic safety functions might not be provided for under all circumstances.:

a) Reactivity control
b) Heat removal from nuclear fuel
c) Capture of ionizing radiation and radionuclides

4.2.1.1 Estimation of safety margin against high wind

The review of resistance of the civil structures for the original design basis levels was executed in 2000. In the years of 2009 - 2010, an independent inspection of the design basis was performed; in particular, the review of parameters for load due to wind was performed in 2010. Calculations of limit load due to wind were executed for all the civil structures subject to review and in some of the civil structure a lower limit resistance than the one corresponding to gust wind with the time of re-occurrence of 10,000 years was discovered.

The consequence of extreme wind may be a loss of the off-site power supply and reduction of the ability to remove heat to the atmosphere through ESW.

Performance of the reactivity control safety function is provided even in case of extreme conditions caused by other climatic incidents and no risk relating to failure to perform this safety function has been identified.

With respect to the fact that the ESW safety system has not an independent system of heat removal to the atmosphere and is connected to the cooling towers, on which overflow for the ESW cooling is executed, the incident of extreme wind, due to the existing resistance of the CHV shell, might result in the reduction of the ability to remove heat through ESW to the ultimate heat sink for all the four units EDU at the same time.

In case of extreme wind it is necessary to take into account a possible failure of the 400kV and 110kV grids (the resistance of the lines is designed for a gust wind not exceeding approx. 38 m/s), which means with transfer to electrical emergency power systems. The loss of the possibility to transfer heat from ESW to the ultimate heat sink might result in the increase of the ESW temperatures and thus in aggravated cooling of DG with possible risk of failure thereof and the follow-up transfer up to the SBO condition (see chap. 5).

Possible damage to CČS roof as a consequence of extreme wind might affect the ESW pumps operability as well. With respect to the location of both CČS as well as with respect to the ESW system configuration, however, it is not likely that the EDU location might lose all 2 x 12 ESW pump at the same moment due to extreme wind. For this reason, the possible risks will not be worse than the aforementioned ones.

A fall of the turbine building´s roof structure due to extreme wind or snow might also means certain risks with respect to the fact that in the turbine building there is safety-related equipment (the reactor after-cooling systems, SG emergency feed water supply, ESW piping, full-pressure steam piping, etc.). Upon the least favourable development of accident, this might result in similar risks as mentioned above as well (possible loss of ESW, loss of DG cooling).

Possible impacts of the fall of the reactor building´s roof have not been analysed in detail yet. Possible damage to fuel in the reactor or BSVP after a loss of the reactor hall´s roof´s integrity due to extreme wind is highly unlikely.

The identified risk lies in the possible loss of ability to provide for heat removal to the ultimate heat sink from all 4 reactors and all BSVPs and in the subsequent loss of ability to monitor the condition of the technological equipment after the capacity of the accumulators has been discharged.

According to the executed analyses, damage to AZ might occur in approx. 9 hours as of the SBO occurrence if no actions are performed. If the initial condition corresponds to a shut-down unit in mode 6 (residual power for 10 days after shutdown was assumed in the calculation), the loss of PC will occur in 8-10 hours and exposure of AZ will occur in 16-20 hours after the loss of circulation in II.O on the conservative understanding that no actions were performed.

For the solution of the extreme wind event, the procedure titled "Destruction of the cooling towers and the 400kV and 110kV lines" has been prepared. This procedure has not been verified in full according to the executed analyses. As it follows from the executed analyses, full utilization of the strategies mentioned in this procedure, technical modifications are necessary (construction modifications).

4.2.1.2 Estimation of safety margin against heavy snowfall and ice

Based on the executed review of load due to extreme weight of snow, an erroneous original calculation of load applied on the roof of the turbine building executed in 2000 was discovered. The new recalculation of resistance of the turbine building was executed in 2010 and the calculated limit resistance of the roof of the turbine building is only 0.95 kN/m^2, which means that the value is lower than that corresponding to the level of the hundred-year snowfall.

The recalculation of resistance of the CČS structure´s and the reactor hall´s roof to load due to extreme snow has not been finalized yet

The effect of ice in the water-management civil structures has been analysed as well. In the water-management civil structures with free surface of water within the EDU´ premises, such amount of ice that would put the operation thereof at risk cannot occur due to extremely low temperatures even in the event that some of the units is shut-down.

A fall of the turbine building´s roof structure due to extreme snow might also mean certain risks with respect to the fact that in the turbine building there is safety-related equipment (the residual heat removal system, PG emergency feed water supply, TVD piping, main steam piping, etc.). Upon the least favourable development of accident, this might result in similar risks as mentioned in the previous chapter (possible loss of TVD, loss of DG cooling).

In case of extreme snow, an immediate effect of extreme weather is not concerned. Therefore, it is possible by means of simple organizational or technical provisions (continuous removal of the fallen snow, roofs, shelters covering the essential equipment) to eliminate the impacts and to provide for the performance of safety functions. However, no procedures and emergency plans are available for the execution of such precautionary activities.

4.2.1.3 Estimation of safety margin against maximum temperature

For the conditions of extremely high temperatures, operation at the temperature of 46.2 ℃ for a period of 6 hours a day is considered. If, under these conditions, breakdown of the off-site grid (400kV as well as 110kV) occurred upon regulation to the unit own consumption, the ČS Jihlava power supply will remain operable and it is possible to supply water from the reservoir to the ESW tanks, which is sufficient for long-term cooling to reach temperatures lower than 33 ℃. In addition, it would be possible to maintain the cooling stations operable and to cool essential coolers of the electro and I&C rooms using chilled water.

Only in case, the breakdown of the grid is combined with failure of regulation of all the units to own consumption, the units would switch to the electrical emergency power systems (DG) and would lose the possibility to make up cool water from the ČS Jihlava. Even after switching to spraying on CHV it would not be possible to maintain the ESW temperature lower than 33 ℃ on a long-term basis, which the va lue relevant for the technical conditions prescribed for a long-term operation of DG (at full output) and the technological condenser. It is obvious that if the ESW temperature is higher than 33 ℃, failure of DG does not occur. Continuation of the DG operation would be possible provided that the output is reduced in a relevant manner so that the temperature of lubrication oil (approx. 60 ℃) and the temperature of the internal circuit coolant (83 ℃) are maintained.

In this mode, it is necessary to maintain the DG systems in a safe condition. High temperature in the environment of the DG box might be a problem as well - as there is no ventilation system. The DG functionality depends not only on cooling water and lubrication, however, on the function of the other systems (generator, excitation, regulation…), which are located in the DG boxes and which might by gradually exposed to temperatures exceeding 60 ℃.

Cases of extreme temperatures are not classified in the category with immediate effect of extreme weather. However, no procedures and emergency plans are available for the execution of the aforementioned precautionary activities. In case of occurrence of extremely high temperatures on a long-term basis, gradual preventive shutdown of the units might be expected.

4.2.1.4 Estimation of safety margin against minimum temperature

During operation at low temperature on a long-term basis, which for EDU has been determined at -35.8 ℃ for a 5-day period with the time of re-occurrence of 10,000 years, all influences and possibilities having a positive impact which are available in the power plant were taken into account in the review. As it followed from the result of the review, the systems of heating and protection against freezing are sufficiently dimensioned and operationally secured to be able to cover the needs of heat under the conditions of extreme coldness.

The effect of ice in the water-management civil structures has been analysed as well. In the water-management civil structures with free water level in the EDU premises, occurrence of the quantity of ice due to extremely low temperatures which might endanger the operation of the civil structures concerned is eliminated.

Upon occurrence of the extremely low temperature event, fulfilment of the safety functions is secured.

4.2.2 Conclusion on the adequacy of protection against extreme weather conditions

Upon occurrence of extreme wind with the time of re-occurrence of 10,000 years, the safety function consisting in residual heat removal might be put at risk. The main cause is the fact that in the TVD system, no forced-draught cooling towers were installed and the main cooling ones fail to have sufficient resistance to extreme wind. In addition, insufficient resistance of some safety-important civil structures upon occurrence of extreme wind has been discovered, however, more detailed impacts on the equipment located there have not been analysed yet. Possible damage to fuel in the reactor or BSVP after a loss of the reactor hall´s roof´s integrity due to extreme wind is highly unlikely.

The most serious impact of extreme weight of snow might be a fall of the turbine building´s roof, which may lead to a loss of the safety systems located in the turbine building. The most serious impact might be outage of the TVD system, which may result in putting the function of long-term removal of residual heat at risk. The aforementioned applies provided that preventive removal of snow from the roof of the turbine building failed. Some partial deviations of the actual resistance of the selected buildings from the required values of resistance due to application of extreme load are solved by the design of completion of seismic qualification for safety-important equipment and civil structures which is being finalized. At the present time, review analyses are under progress to re-prove sufficient resistance to the effects of climatic extremes for all civil structures, systems and components which provide for the performance of basic safety functions.

The assessment of extreme climatic incidents was reduced only to the scope of safety relevant civil structures and equipment located therein. Therefore, it is necessary to assume that in particular an event such as extreme wind and extreme snow might result in damage to civil structures providing form auxiliary services. Such events might even cause the location´s isolation and its inaccessibility for a period of several days.

The FB civil structure was not subject to the assessment with respect to extreme climatic incidents (extreme wind, extreme snow, earthquake). Therefore, it is not known whether the

FB civil structure might be damaged as a consequence of climatic incidents or not. At present, analyses of resistance of the HZSp´s building are being performed.

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

The purpose of the proposed provisions is the consolidation of the in-depth defence in case of extreme climatic incidents. The opportunities for improvement of the in-depth defence are provided in the following table. The table contains also areas for which it is necessary to execute supplementary analyses as at the time when the review was executed they were unavailable. A more detailed description of the corrective actions is provided in the annex.

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
Diverse CHV	Implement provisions for diverse means of the ultimate heat sink (to CHV)	II	PSR, Under implementation
Regulations	Prepare the operating procedure for extreme events (wind, temperature, snow)	I	
Regulations	EDMG guidelines for the use of alternative means	II	
Staff	Provision of sufficient staff after occurrence of extreme events	I	
Analyses	Resistance of civil structures (HZSp, CČS, HVB, etc.) to extreme conditions	I	Under implementation
Analyses	Preparation of the methodology for assessment of external influences, verification of the executed analyses, possible technical provisions	II	

5 Loss of electrical power and loss of ultimate heat sink

For clear understanding of the following text, it is necessary to get familiarized with the content of chapter 1.3 describing the technological systems necessary for the provision of fulfilment of safety functions, and in particular chap. 1.3.5 and 1.3.6 describing power sources of the Dukovany NPP.

5.1 Nuclear power reactors

5.1.1 Loss of electrical power

The electrical systems of EDU are designed in a manner allowing to meet the respective requirements imposed on the machine and nuclear components and to respect the properties of the electrical grids outside the NPP, namely in particular with respect to the EDU´s operational safety and power generation.

Provision of safety in case of loss of power supply is resolved on a design basis with a high level of mutual independence of the operational as well as stand-by power supply units for own consumption together with redundancy of the so-called electrical emergency power systems (ZN), which power supply the safety-relevant systems and components and which dispose of own stand-by power supply units.

The power distribution grid for own consumption of EDU is power supplied from diversified operational, stand-by and emergency power supply units.

The operational power supply for own consumption is designed on the basis of the individual units. The unit own consumption power supply can be provided in two ways due to the application of the generator breakers:

- by means of power supply from the 400kV off-site grid (under no-power mode and in outage)
- by means of own TG (during power operation of the respective unit).

The point of connection of the EDU to the 400kV off-site grid is Slavětice switchyard, to which the output of all the NPP four units is led-out and which provides for the operational power supply for own consumption (PNVS). Own consumption of each unit is provided on a standard basis from the pair of the on-load tap-changing transformers connected on the branch from the output lead-out line of each TG. On unit 1, in addition, an on-load tap-changing transformer having the same parameters is installed for power supply of aggregated own consumption of the entire plant. These power supply units are used during normal operation as well as abnormal operation as well as under accident conditions provided that the link to the 400kV grid or power supply from the TG remained functional.

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

Insular operation of EDU:

The EDU units are able to work in the isolated part of the grid, under the so-called insular operation. The insular operation is activated by deviation of the grid frequency ± 200 mHz determined by the FREA16 frequency relay. The TG control system contains a specific insular control (proportional frequency control) whose primary function is to maintain the frequency in the insular grid. However, after transition to “island”, the nuclear source has its specifics relating to the provision of nuclear safety. From the point of view of JB, no sudden changes of the reactor power occurring during operation in the insular mode are desirable. Sufficient amount of "promptly available" power is provided by the transition of the steam dump to condenser to the margin control. The margin is created automatically by the “Insular Operation” automatics and is set automatically up to 20% opening of each train of the steam dump to condenser. For scenarios relating to extensive relief of TG, the design contains the automatics for assessment of high acceleration of TG and a pulse derived therefrom to the TG´s hydraulic control accelerator (the so-called runout control).

A specific guideline within the framework of the operating procedure for abnormal conditions (P002b) has been prepared for insular operation. The range of frequency of the insular grid, within which the EDU units are able to work on a long-term basis is limited by the 2-degree setting of the FREA16 frequency relay - for frequencies lower than 47.9 Hz or higher than 52.5 HZ (with a delay of 25 s), automatic disconnection of the unit from the insular grid and transition to own consumption occurs.

The measurements in the course of the event that occurred on 3 August 2006 confirmed the high quality of the TG revolutions´ control through the insular control as well as of the other functions to support operation in the insular operation mode. In addition, the EDU´s units worked in a real, slightly surplus island upon disintegration of the UCTE grid into three isolated units on 25 July 2006.

The ability of insular operation of the EDU´s units is certified as auxiliary service for the Czech Republic´s grid operator. Detailed information are contained in the ICE-OP-28,29,31/2007 certification report for RB 1, 2, 4; to RB 3, the ICE-OP-38/2009 certificate is applicable. At the present time, a new certification measurement has been executed at RB 1 and a new certificate is going to be issued.

The stand-by power supply unit for own consumption are two 110kV switchyards, Slavětice and Oslavany, both of which are equipped with mutually interchangeable 110kV supply lines. Their supply is possible by means of multiple 400/110kV transformations and 220/110kV supply lines separated from each other spatially as well as electrically from different directions and nodes of the power grid. The connection to the 110kV off-site grid, from which the stand-by power supply for own consumption (RNVS) is power supplied, is from the 110kV switchyards Slavětice and Oslavany; from the point of view of power supply units, the 400/110kV (220/110kV) transformation in the Slavětice, Sokolnice and Čebín is concerned. The stand-by power supply for the twin unit (HVB) is provided by means of two RNVS transformers which are connected to the 6kV unit switchgears through the stand-by bus bars. Using the breakers on the 6kV side, it is possible to disconnect the RNVS system of HVB I and HVB II. This allows for mutual redundancy of the RNVS power supply units for both the HVBs. Under no-power modes, some of the operational or stand-by power supply units may be unavailable on a long-term basis due to the regular maintenance execution.

Emergency (accident) power supplies refer to three automatically quick-starting Diesel generators for each unit (DG, U_n = 6.3 kV, Pn = 2.8 MW, Sn = 3.5 MVA) and the unit accumulator batteries SNZ1, 2 and 3, (voltage 220 V, capacity 1,500 Ah, and voltage 48 V, capacity 2 x 246 Ah). The power flows of the uninterrupted power supply consumers are provided by rectifiers and inverters.

For power supply of the safety systems, each unit is equipped with 3 systems of the electrical emergency power system, designated as ZN 1, 2, 3 (3 x 100 %). To secure the necessary degree of redundancy, these ZN 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.

For power supply of the part of the systems related to nuclear safety and for power supply of the systems irrelevant from the point of view of nuclear safety, which however provide for general safety of persons and expensive equipment, ZN 4 is used, which is designed as two sub-systems mutually redundant based on the 100 + 100 % principle. The emergency power supply of each of the systems is the station accumulator battery with capacity of 2,000 Ah, 220 V and the set of uninterrupted power supply. The SZN 4.1 is connected to SZN1; SZN 4.2 is connected to SZN2.

A detailed description of the AC power supplies is provided in chap. 1.3.5 and the description of the DC uninterrupted power supply units is provided in chap. 1.3.6.

5.1.1.1 Loss of off-site power

A loss of power supply may affect one or several units of EDU. Power operation of the unit is characterized by a higher design basis resistance to loss of power supply (additional barrier of in-depth defence) than in case of refuelling outage. Nevertheless, a loss of power supply on all the units at the same time is regarded as the most adverse case from the point of view of the safety provision. As far as the possible configuration of the available equipment is concerned, the most conservative case is the condition under which some of the units is in outage.

5.1.1.1.1 Design provisions

Loss of the off-site supply (e.g. upon disintegration of the grid accompanied by the concurrent loss of the 400 kV as well as the 110 kV switchyards, will not result in an automatic switch to the emergency power supplies at the unit´s power operation.

In case the unit is disconnected from the 400kV off-site grid due to external reasons (i.e. failure in the 400kV power distribution grid, which is not related to failures of the nuclear unit and its 400kV line to the nearest 400kV switchyard Slavětice), a design basis regulation of the TG occurs so that the output thereof covers the unit own consumption. The on-load tap-changing transformers power supply the 4 unit switchgears of the 6kV power supply for own consumption, from which the main actuators of circuit I and circuit II are power supplied, including the switchgears of the 6kV electrical emergency power system, which power supply the actuators of the safety systems.

If the aforementioned steps fail to be executed (the units in outage, TG do not work or neither of them has been regulated or is subject to shutdown), the section of the switchgears of the 6kV power supply for own consumption are automatically switched to the 110kV stand-by power supply units (an aggregated automatic switch to the stand-by power supply is executed). The diesel generators (DG) in this case are not started, the accumulator batteries provide for continuous power supply of the direct-current power supply distribution systems. In case of failure of the automatic switch to the standby power supply units, a loss of the operational as well as the stand-by power supply units of the respective unit would occur, which means the so-called total loss of power supply for the unit own consumption (LOOP).

A loss of the operational as well as the stand-by power supply units of the respective unit will result in a decrease of voltage in the switchgears of the 6kV electrical emergency power systems. Switching off the sectional breakers will result in the ZN disconnection from the normal power supply grid. In this case, the sectional breakers will disconnect the switchgears of the 6kV unsecured electrical emergency power systems and will start all three DG. After the connection of DG to the switchgears of the 6kV ZN, the safety-relevant actuators are activated gradually and automatically based on the program of gradual load application depending on the unit mode (hot or cold state).

Under this mode, the accumulator batteries are recharged in the standard mode and provide for uninterrupted power supply of the direct-current power supply distribution systems. Recovery of recharging of the accumulator batteries will occur after the DG´s connection to the SZN 6 kV switchyard, which means within 15 sec. maximum, within 10 sec. in reality.

A loss of the NPP off-site power supply units, means no imminent risk to the fulfilment of any of the basic safety functions:

a) Reactivity control.

b) Heat removal from the nuclear fuel.

c) Capture of ionizing radiation and radionuclides.

Under the mode of the off-site power supply loss, the EDU units can be maintained on a long-term basis either in a safe state or they can be after-cooled until cold state or maintained safely under the mode of outage (power supply of all necessary mechanical systems as well as SKŘ systems is provided) on condition that at least one of the three DGs is activated on each unit.

In case, the unit is on power operation during LOOP, the reactor is scrammed due to the RTS signal and all the main circulation pumps (HCČ) are scrammed as well. Removal of residual heat from core proceeds under the mode of natural circulation, by removal of steam from SG through the turbine bypass steam dump to the atmosphere. Water make-up to SG is provided by means of two auxiliary feed water pumps (AFWP), which pump water from the feed water tank (FWT), the make-up of which proceeds by means of the 1 MPa demineralised water pumps from the 3 x 1,000 m3 tanks. Alternatively, water make-up to SG is possible by means of the so-called emergency feed water pumps (EFWP), which would pump water from the 3 x 1,000 m3 tanks directly to the selected SG in case of failure of the SG make-up by means of AFWP1, 2.

If, under LOOP, the unit is in shutdown state with water-water after-cooling, water make-up to SG is provided by means of the after-cooling pumps (DČ) in a closed circuit. Alternatively, it is possible to provide for water make-up to SG by means of EFWP, which would pump water from the 3 x 1,000 m3 tanks directly to the selected PGs. With the open reactor and low level of coolant at the beginning of Mode 6, however, activation of both EFWP is necessary to eliminate the loss of natural circulation.

Cooling of the spent fuel storage pool (SFP) is realized by means of two cooling circuits. Each of the cooling circuits includes a circulation pump a heat exchanger. The heat exchangers are cooled by essential service water. The SFP cooling system pumps as well as the ESW pumps are power supplied from DG and are activated automatically under the program of ELS.

All the aforementioned loads are power supplied from DG (3 x 100 %) in compliance with the VVER concept when each ZN power supplies all the actuators, which are necessary for the provision of safety functions under the respective mode. The only exception is the aforementioned SFP cooling system, which is covered by only two systems (ZN 1 and 3).

5.1.1.1.2 Autonomy of the on-site power sources and provisions taken to prolong the time of on-site AC power supply

In compliance with the basic concept of the machine-nuclear part (3 redundant and independent divisions of safety systems), there are also 3 redundant and independent secured power supply systems available (3x100%). Each of these SZN is a supporting system for the safety systems of the respective division.

- The emergency AC power supply sources for the SZN of safety systems comprise three independent (systems) DGs, which are connected to the respective 6kV switchyards of the safe (uninterruptible) power supply.
- The DC emergency power supply sources are accumulator batteries that are permanently connected to the respective switchyards. In case of loss of working and standby power sources of the unit and after the DG´s connection, the accumulator batteries are recharged in a standard mode and provide for uninterrupted power supply of DC distribution systems.

The supply of fuel oil in the operating tank for each DG is sufficient at least for 6 hours; in case of the fuel oil repumping from the storage tanks, operation of one DG can be secured for at least 144 hours. Other fuel for DG could be obtained by repumping from the storage tanks of the other DGs. While considering a long-term operation of always just one DG on each unit, putting the reexpedition pumps into service would mean to have fuel for a period of 18 to 21 days at the disposal without the necessity to supply EDU with fuel oil (for details, see 1.3.5.3). The steady-state load applied on SZN 1, 2, 3 is lower than the nominal output of DG (2.8 MW). The only limiting factor concerning a long-term state of the off-site power supply loss could be the fuel oil reserve. As mentioned above, each of the safety DGs is provided with a reserve of fuel oil for minimum 6 to 7 days of operation without the necessity to refill fuel from external sources.

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

5.1.1.2.1 Design provisions

Upon loss of working as well as standby power supply combined with loss of the unit´s emergency power supply (DG), the following power supplies can be available to secure AC power supply:

Use of the EDU on-site equipment:

The use of the autonomous alternating current power supply units in EDU is expected and the possibility of easy connection thereof through the 6 kV standby bus bars, which is described in EOPs or AOPs respectively. In case of the SBO occurrence on only some of the EDU units and in case at least two DGs are under operation on one of the remaining units, it is possible to provide for power supply of the SBO-affected unit from the DG operating on the other unit.

Therefore, there are the following options available:

- Voltage recovery from the EDU unit by means of the 6kV stand-by bus bar.
- Voltage recovery from DG by means of the 6kV stand-by bus bar

Use of the 400kV or 110kV lines:

If, after events resulting in SBO, the possibility to earmark the 400kV or110kV power supply lines for EDU remains available, the unit own consumption power supply of the selected units would be provided primarily from the nearby Dalešice and Vranov hydraulic power plants in compliance with procedures specified in AOPs. The prerequisite of the aforementioned, however, is the possibility of communication with the respective off-site workplaces (EDA, VRV, switchyards Slavětice, dispatching centres of ČEPS, E.ON) Recovery of power supply from VE Dalešice (4 x 112.5 MW) or from VE Vranov (3 x 6.3 MW) has been tested on a repetitive basis (in 2004 and 2010) and its availability has been proven during the testing.

After execution of the SBO analyses, the Dalešice hydraulic power plant was selected as the main external AAC power supply source and this function was tested in practice. The Dalešice hydraulic power plant (power 4 x 112.5 MW) is able to start from black. The ability to provide for power supply within 30 minutes (along the 400kV line) or within 60 minutes (along the 110kV line) has been tested.
For further details, see chap. 1.3.5.1.2.

If SBO occurs on the unit in hot state, SI EDU shall impose the state of EXTREME EMERGENCY, which, according to the Code of the Transmission Grid of the Czech Republic, means the necessity to supply power from the off-site grid to the affected unit within 1 hour. If SBO occurs on the unit in semi-hot state, the state of DANGER shall be imposed with the necessity to supply power from the off-site grid to the affected unit within 2 hours.

Recovery of power supply to the safety-relevant equipment (on condition that the respective off-site routes and switchgears are available) is provided by means of the following variants:

- Recovery of voltage from the Slavětice or Oslavany switchyards through the 110kV line
- Recovery of voltage from the 400kV system through the 110kV line
- Recovery of voltage from the 400kV system through the 400 kV unit line
- Recovery of voltage from EDU unit through the 110kV line
- Recovery of voltage from EDU unit through the 400 kV unit line

Power supply for EDU from diverse sources of AC power supply:

- Recovery of voltage from EDA through the 110kV line
- Recovery of voltage from EDA through the 400kV unit line (tested in 2004)
- Recovery of voltage from VRV through the 110 kV unit line (tested in 2010)

All the aforementioned methods of voltage recovery from the off-site power supply units are conditioned by the availability of the necessary segments of the 400kV and 110kV lines

5.1.1.2.2 Battery capacity, duration and possibilities to recharge batteries

The capacity of the accumulator batteries of SNZ1, 2 and 3 is 1500 Ah. For each ZN 4.1 and 4.2, the accumulator battery capacity is 2000 Ah, for details, see 1.3.6.

In case of loss of the operational as well as the stand-by power supply units of the respective unit and DG activation, the accumulator batteries are recharged under the standard mode and provide for uninterrupted power supply of the direct current power supply distribution systems. Recharging will be started upon recovery of AC power supply from the AAC sources which are specified in chapter 5.1.1.2.1.

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

The SBO refers to an accident on the EDU unit characterized by a loss of all the operational, the stand-by as well as the emergency alternating current power supply units of the respective unit - with the unit under power operation, e.g. after failure of the power grid, failure to regulate neither of the two turbo-generators for own consumption and failure to activate power supply from any of the three diesel generators of the respective unit.

An event including a total loss of power supply of the blackout type (SBO) occurring on the EDU unit refers to a beyond design basis accident. The SBO is to be resolved from the point of view of a single unit in case the whole plant is affected by LOOP. The most serious mode is the occurrence of SBO on all the EDU units at the same time. It might occur only in the event that all the undermentioned levels of in-depth defence of power supply failed at the same time:

- External working power supplies - standard power supply from the 400kV switchyard Slavětice
- Internal working power supplies - failure to control any of the turbine generators for own consumption
- External standby power supplies - standby power supply from the 110kV switchyard Slavětice
- External standby power supplies - standby power supply from the 110kV switchyard Sokolnice
- External standby power supplies - standby power supply from the 110kV switchyard Čebín
- Internal standby power supplies - power supply from VS of the twin unit
- All the three redundant emergency AC power supply sources for SZN 6 KV on all 4 units of EDU (which means 12 DGs in total)
- Diverse external AC power supply - the Dalešice hydraulic power plant through the 110kV line
- Diverse external AC power supply - the Dalešice hydraulic power plant through the 400 kV line
- Diverse external AC power supply - the Vranov hydraulic power plant through the 110kV line

During SBO, an immediate failure of the technological systems located in the EDU off-site premises which are not power supplied from accumulator batteries (due to the loss of power supply of the joint unit own consumption EDU) occurs, e.g. the compressor stations (NTKS, VTKS), the pump houses (CČS1,2, ČSJ), the cooling stations and the other auxiliary systems.

The only power supply unit under the SBO mode are the accumulator batteries, or the local UPSs. After capacity thereof has been exhausted, a loss of all measuring and control equipment will occur, including the reactivity control systems (ExCore, InCore), the dosimetric systems, the computer systems, the physical protection systems, the emergency lighting, the fire alarm and detection system, the on-site warning systems (internal radio, hooters), the telephone and other communication systems.

The I&C systems to measure reactivity (ExCore, InCore) and the post-accident monitoring system PAMS are power supplied from the ZN 1, 2, 3 accumulator batteries.

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

Under SBO, discharging of the accumulator batteries occurs as there is no system for their recharging available. The discharge time of the safety systems batteries depends on the current load behaviour in time. The capacity of the accumulator batteries of SNZ1, 2 and 3 is 1500 Ah. The endurance of the accumulator batteries required by the design upon maximum load is minimum 2 hours. While taking into account the current state of the accumulator batteries and the actual load and while considering the reduction of the connected consumers, the reasonable expected endurance can be several times longer. To spare the direct circuit power supply units, it is possible to disconnect all unnecessary consumers, including a part of the emergency lighting. It has been demonstrated through analysis that even the minimum reduction of their load will result in the extension of the time of usability thereof up to 6 - 8 hours, the respective manuals have been prepared in EOPs already. For further, more effective saving of the DC power supplies, it is possible to disconnect all unimportant current consumers, including emergency lighting. Another possibility how to extend power supply of the SZN 1,2,3 current consumers is the use of only one system at the given moment. This more effective saving may effectively extend the time of usability thereof up to 10-20 hours. Another possibility is to use the energy repumped from the batteries of the SZN 4 system. For each ZN 4.1 and 4.2, the accumulator battery capacity is 2000 Ah, but the real endurance of the accumulator batteries without fundamental reduction of the load is app. 6 hours. However, more detailed guidelines and instructions have not been prepared yet. For further details, see chap. 1.3.6.

The current design solution, effective on 30 June 2011, recharging of the batteries in a different manner, for example, from the mobile DCs, was not taken into account. However, such a possibility is technically available.

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

In the EDU´ location, with the exception of portable power supply units of HZSp (see chap. 6.1.3.7), no alternative or mobile AC power sources that could be used to deal with a long-term SBO are available on the site. Nevertheless, there are external power supply sources, the availability and usability of which for the SBO solution have been verified and tested. The Dalešice hydraulic power plant (power 4 x 112.5 MW) is able to start from black. The ability to provide for power supply within 30 minutes (along the 400kV line) or within 60 minutes (along the 110kV line) has been tested. This ability of the so-called blackstart is subject to periodic certification for all 4 generators of the Dalešice hydraulic power plant and in cooperation with the Dukovany NPP, this ability represent the basic method of the power supply recovery after a possible disintegration of the grid in the Czech Republic.

5.1.1.3.3 Competence of shift staff to make necessary electrical connections

For the execution of activities under SBO, the EOPs procedure is determined to resolve handling aiming at providing for heat removal from core as well. Although heat removal from SFP under SBO is not specified in EOPs, it is referred to by means of TSC instructions. As well as the activities necessary to spare the DC power supply units (accumulator batteries) are not specified sufficiently and neither is the uneven distribution of load from individual systems with specific conditions on unit 1.

The activities under long-term SBO shall include consistent conservation of the direct current power supply units. After disconnection of the emergency lighting, it is necessary to replace it with hand lamps, which are necessary for work for example in the switchgears and the other closed spaces as early as from the occurrence of SBO (the emergency lighting is designed for safe leave of the workplace only).

The staff capacity for the early stages of SBO is sufficient, however, in case of a long-term SBO, the staff assignment shall be subject to special regime (altering in exposed workplaces, organization of rest, meals and conservation of available resources).

A priority activity shall be the provision of heat removal from core and from SFP. The SFP cooling would proceed according to the TSC instructions (if available) or instructions issued by Shift Supervisor and Safety Engineer, in cooperation with the fire brigade and the dosimetric service.

Movement of members of the staff under SBO would be restricted by the blocked revolving doors and limited by the radiation situation within EDU, which would be mapped based on manual measurements only after the accumulator batteries have been discharged.

Functionality time of the shelters and the assembly points is determined by the ventilation systems´ operability, the accumulator batteries´ capacity and by the functionality of the information and communication means. In particular the shelters where the staff necessary for interventions is gathering (including EC and TSC) count among the difficult ones. These shelters are power supplied only from the unsecured electrical emergency power systems and under SBO, the TSC and EC technical equipment is not functioning.

In case of failure of the TSFO power supply system from the 6kV switchgear for own consumption, an automatic uninterrupted shift to power supply from the central UPS of the TSFO system will occur. After discharge of the central UPS power supply unit, the terminal equipment of TSFO will cease to function. In addition, the FO EDU operation shall be provided according to the internal FO EDU documentation.

5.1.1.3.4 Time available to provide AC power and to restore core cooling before fuel damage

Under SBO without action taken by the staff, residual heat removal from core is executed in the mode of natural circulation by means of removal of steam from SG through steam dump to atmosphere, which are power supplied from the accumulator batteries and which can be operated manually as well (on spot). Water supply to SG is however interrupted and a gradual decrease of level in SG occurs with the follow-up reduction of the efficient heat-transfer surface. The ability to remove heat within II.O is thus gradually reduced. If no activity (described further in this chapter), which is imposed by the EOPs procedures, is performed, the heat removal from the core by means of natural circulation through SG fails, the temperature and the pressure in I.O. would increase with the follow-up opening and coolant removal through PORV or/and SV PRZR. This will stop the increase of temperature in I.O on a temporary basis and the decrease of level in the reactor is suspended temporarily by passive discharge of hydroaccumulators. This would mean uncompensated loss of primary coolant and increase of parameters in the SG box at the same time.

The EOPs procedures, however, provide the BD´s staff with sufficient instructions to enable heat removal from I.O in the Feed and Bleed mode on the part of II.O. The time allowance to eliminate the loss resulting from heat removal from I.O is approx. 4 hours. If the SBO condition occurs, the staff, in accordance with the EOPs procedure, will start depressurization of PG I.O using PSAs. The PSAs are power supplied from the accumulator batteries and it is possible to control them mechanically (on the spot) as well. With SG depressurization to approx. 0.7 MPa, spontaneous (gravity) outpour of feed water will occur from FWT (2 x 150 m^3) to SG with the follow-up temporary recovery of heat removal. Heat removal can be provided in this mode for a period of approx. twenty hours as of the SBO occurrence.

After depletion of the NN reserves, another possibility how to proceed with heat removal in the Feed and Bleed mode on the part of II.O (described in EOPs) by make-up of demineralised water directly to SG in an alternative manner using the fire water mobile pumps (with pressure at the pump delivery of 0.8 -1.2 MPa, flow rate120-150 t/h). Within the framework of the EDU´s upgrade with respect to the SBO consequences, connection points enabling interconnection between the fire-suppression equipment and the technological equipment were implemented on all the units. The alternative way of SG make-up is described in EOPs and has been tested in practice several times and the capacity of this equipment to secure basic safety functions has been verified. Upon black-out affecting all the four units of EDU, however, the capacity of the necessary fire-suppression equipment may be a certain limitation (as the emergency plans for the PG supply in two units by means of one pump at the same time have not been prepared yet). For demineralised water make-up by means of optimal flow rate, the existing reserves of demineralised water in the 3 x 1,000 m^3 tanks for each twin-unit are available, which, according to the respective analyses, would be sufficient for 72 hours for all the 4 units. Together with the use of the coolant reserves in FWT, the coolant reserve for the SG make-up of all the four NPP units is sufficient for approximately 4 days.

In case of operation during outage with low level of coolant in the reactor, the critical phenomenon is the loss of natural circulation. If the SG make-up is not activated within approx. 4 hours, the loss of natural circulation due to decrease of level in the reactor under the hot leg nozzles will occur. Subsequently, the core cooling by the SG make-up only would be ineffective. To prevent damage to the core even in case of a long-term SBO in the early stage of accident, it is possible to use coolant from the hydraulic accumulators to recover the level in the reactor. When reactor is opened very efficiency is to keep level of boiling water in open reactor using gravity make up from bubbler trays. Possibility to maintain water level in reactor is app. 12 days.

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

The power supply sources for EDU provide for sufficient design basis robustness as well as degree of safety provision upon loss of the off-site power supply. They are designed with a high degree of independence, mutual back-up and redundancy (see the working and standby power supply for own consumption together with the emergency AC and DC power supplies, the so-called safe (uninterruptible) power supplies - SZN which supply power to systems and components important from the point of view of safety).

There is a higher design-basis resistance to the loss of power supply if the unit is in the mode of power operation (sufficient in-depth defence barriers) than during refuelling outages. Nevertheless, loss of power supply on all the units at the same time is regarded as the most adverse case from the point of view of the safety provision.

In the location, there are in total 12 standby AC power supply units (DGs) available. Under the mode of the off-site power supply loss, the EDU units can be maintained on a long-term basis either in a safe state or they can be after-cooled until cold state or maintained safely under the mode of outage (power supply of all necessary mechanical systems as well as SKŘ systems is provided) on condition that at least one of these DGs is activated on each unit. For each DG, the fuel oil supply is sufficient for 6 to 7 days without the necessity of external fuel refilling.

From the point of view of the design basis resistance to LOOP, there are, both in case of power operation of the unit, and in case the unit is shutdown, minimum two independent ways of automatic provision of power supply of the double- up to triple-redundant systems necessary for the fulfillment of safety functions (by means of the 220MW alternating current generators after the TG regulation for own consumption or by means of the redundant DGs). In case of low-output or shutdown states, it is not possible to apply the TG regulation to the unit own consumption. A long-term operation under the conditions of the off-site power supply loss is guaranteed for a period exceeding 72 hours.

In case of complete loss of the alternating current power supply units (SBO), the only power supply units to power supply the safety systems and the safety-related systems are the direct-current uninterrupted emergency power supply systems (accumulator batteries). Without the respective DG operation, the accumulator batteries are not recharged and the time available prior to their discharge is in the order of hours up to tens of hours depending on the currently applied load. This time is sufficient for recovery of power supply of the EDU´s VS units from the neighbouring Dalešice or Vranov hydraulic power plants.

Consequences of a long-term SBO may be as follows:

If either alternating current power supply or heat removal by means of diversified system is not provided until uncovering of fuel in core or SFP, then damage to either fuel in the core or in SFP may occur.

To prevent the aforementioned development of SBO from the point of view of the technological equipment, there are sufficient alternative sources of coolant available for long-term cooling of AZ as well as BSVP without power supply. In accordance with EOPs, during SBO, residual heat removal from AZ would proceed in the mode of natural circulation by means of steam removal from PG though PSAs in the mode of the so-called secondary Feed and Bleed. With SG depressurization to approx. 0.7 MPa, spontaneous (gravity) outpour of feed water will occur from FWT (2 x 150 m^3) to SG with the follow-up temporary recovery of heat removal. Heat removal can be provided in this mode for a period of approx. twenty hours as of the SBO occurrence. Another time reserve for power supply recovery. The alternative way of SG make-up described in EOPs and has been tested in practice several times and the capacity of this equipment has been verified. There are connection points available which enable interconnection between the fire-suppression equipment and the technological equipment. The existing reserves of demineralised water in the 3 x 1,000 m^3 tanks for each twin-unit are available, which with feed water with FWT would be sufficient for feeding SGs for approximately 4 days for all the 4 units. As an alternative, it could be possible to use water from the over flowing pools of CHV.

Heat removal from BSVP during SBO would proceed in the mode of coolant boiling in BSVP. There is possibility to keep level of boiling water in SFP using gravity make up from bubbler trays. Possibility to maintain SFP cooling is app. 13 days.

If discharge of the batteries occurs, the loss of the I&C systems´ functionality would result in the loss of control over the systems and components as well as loss of ability to communicate essential parameters.

After discharge of the batteries, the loss of lighting and the failure of the TSFO technology would occur with the subsequent extension of the time necessary for the execution of the work concerned due to aggravated orientation and movement of the staff.

The loss of operability of the technical means designed for communication, warning and information of the staff. The aforementioned represents a threat to communication between the control centre and the persons executing interventions, including communication with the off-site control centre and the respective bodies of the state administration.

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

The goal of the short-term measures proposed further on is to eliminate the most critical risks by means of consolidation of the in-depth defence levels during initiating events exceeding the framework of the existing design (earthquake, floods, extreme conditions, human activity consequences, etc.), which may result in the loss of ability to fulfil the safety functions during SBO:

1. Propose and implement the alternative means of the alternating current power supply of the existing equipment to provide for cooling and heat removal from the core and SFP, including the possibility of their connection to the existing power supply distribution system.

2. Propose and implement the diversified means to provide for cooling and heat removal from core and SFP, including the possibility of their connection to the existing technological systems.

3. Propose and implement the alternative means to provide for direct current power supply and cooling of the I&C systems necessary to provide for the monitoring of condition and control of the selected components.

4. Propose and implement the alternative means to provide for the activities and functional communication (on-site as well as off-site) of the staff.

The opportunities for improvement of the in-depth defence are provided in the following table. The table contains also areas for which it is necessary to execute supplementary analyses as at the time when the review was executed they were unavailable. A more detailed description of the corrective actions is provided in the annex.

Opportunity for	Corrective action	Deadline	Note

improvement		(short-term I / medium-term II)	
Power supply of category I	Provide for sufficient power supply of the ZN systems of cat. I and the classified current consumers of ZN cat. II	II	
Heat removal from AZ through II.O	Provide for sufficient source of the PG volume control	II	
Heat removal from AZ through I.O	Analyse the possibility of alternative make-up of the reactor using a pump and a new pipeline	II	
Regulations	Prepare the procedure for power supply recovery after SBO on all the units	I	
Regulations	Prepare the procedure for PG volume control for all the four units by means of fire-suppression equipment	I	
Regulations	Filling the open reactor and BSVP by gravity from the XL conduits	I	Under implementation
Regulations	EDMG guidelines for the use of alternative means	II	
Analysis	Analysis of the discharge time of the accumulator batteries upon application of the controlled decrease of load, completion of PP, modification of connection and operation of the emergency lighting system	I	Under implementation
Sheltering of staff and communication	Provide for an alternative power supply source for shelters and telephone exchanges	II	
Staff movement	Provide for and alternative power supply source to supply power to TSFO	II	
Staff	Provision of sufficient staff in case of long-term SBO	I	
Emergency preparedness	Provide for the functioning of the emergency response teams in case of HŘS´s unavailability	I	

5.1.2 Loss of the ultimate heat sink

5.1.2.1 Design provisions

The ultimate heat sink in case of the EDU units 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:

a) Through the secondary circuit using the condensation and circulation cooling water system - under normal as well as abnormal operation in the mode of power operation, the TG activation and scram and in the accident mode after the reactor scram on condition that the operational or the stand-by power supply units are provided. This way does not provide for the reactor transition to the cold state.

b) Using the after-cooling system with heat exchange to essential service water (ESW) - under normal as well as abnormal operation and under accident conditions, transition of the reactor to the cold state (approx. 50 ℃ in the cor e as well as in SFP) is possible.

c) By means of direct steam extraction from SG to the atmosphere with the simultaneous SG make-up with feed water - under abnormal or accident operation; the reactor transition to cold state (after-cooling to max. approx. 110 ℃) i s not possible.

d) The alternative way of after-cooling, by the Feed & Bleed method on the primary circuit (SV PRZR + ECCS) with heat removal to essential service water - only under accident conditions in case of loss of the secondary circuit equipment. From the point of view of the ultimate heat sink, this method is equivalent to heat removal from the after-cooling system to the ESW system (similar to b).

Heat from the spent fuel storage pools is removed by the ESW system.

Heat removal from I.O through the SG secondary side is provided after the reactor scram while the unit is maintained in hot state or while the unit is being after-cooled. It is provided by throughflow of feed water to SG (the feed water system or the emergency feed water system) and by removal of steam from SG to the TG condenser or to the atmosphere. Heat removal from the TG condensers to the circulation cooling water system is not subject to further assessment as it may be unavailable (systems non-essential from the point of view of safety are concerned).

For cooling I.O. to reach cold state, for removal of heat from spent fuel in SFP and for removal of heat from consumers of the safety systems and the nuclear safety-related systems, the essential service water system is used, which removes heat to the atmosphere as the ultimate heat sink. All the three ESW systems are under operation at the same time (3x100 % redundancy).

Therefore, the assessment of the ultimate heat sink loss further focuses in particular on the loss of the ESW system Heat removal through the secondary circuit can be provided by a combination of multiple options from the point of view of solution redundant from the point of view of the systems configuration. The systems participating in heat removal are available even in case of power supply limitation to operation of the stand-by power supply units.

The loss of the ultimate heat sink is assessed from the point of view of the entire location. The ESW system has the twin-unit layout with operation of the pumps of both the units into the common pipeline within the framework of the division. The loss of ESW will result in lesser risks for heat removal from core in modes, in which the reactor is sealed (all modes of the unit with the exception of a refuelling outage) as it is possible to rely on the physical principle of natural circulation of coolant in the closed I.O with after-cooling in SG from where it is possible to provide for heat removal by some of the aforementioned ways.

During refueling (or in state in which the reactor is not sealed up), heat removal proceeds through SG in the water-water mode. The loss of forced ESW throughflow then means similar risks for fuel deposited in the open reactor as for the fuel deposited in the SFP.

5.1.2.2 Loss of the primary ultimate heat sink

The loss of the ultimate heat sink cannot put the reactor scram, either automatic or manual, at risk. Formation of the scram-activating concentration can be provided by the operation of the emergency boration system pumps (they do not need TVD for operation).

The ultimate heat sink used by EDU - the atmosphere - cannot be lost. Residual heat removal from fuel deposited in the reactor or in the storage pools, or from the SFP consumers and the safety systems to the ultimate heat sink is based on the physical principal of passive heat transfer from the auxiliary media to the atmosphere. The loss of the ultimate heat sink can be therefore assessed as the loss of the heat transfer ability only, which means the loss of the systems providing for flows of media for heat transfer between the heat sources and the atmosphere.

The loss of the ultimate heat sink as such, cannot endanger the containment integrity in case of modes with the closed reactor. In case of loss of the ultimate heat sink with open reactor, there is no other barrier to capture ionizing radiation and radionuclides available and there is a risk of possibility of leakage thereof to the reactor hall or even outside NPP. See 5.2.2.2.

Impossibility of cooling water make-up for EDU and loss of heat removal from TVD to the atmosphere

The EDU designs consider a loss of raw water make-up to the cooling water volume. In this case it is assumed that heat removal from the ESW system can be provided to the atmosphere through the CHV overflowing using the EDU water reserves. In case, the unsecured power supply unit is available, it is possible to use the reserve of coolant in the decantation tanks in the amount of approx. 5 x 2,000 m3 and the raw water supply in the gravity distribution reservoirs with the volume of 4 x 2,000 m3 to compensate the losses of ESW by evaporation.

The loss of the CHV function (TVD overflow on CHV will not be available) in case it is possible to make up raw water from ČS Jihlava is not critical for EDU provided that the ability of the ESW system to supply water to the consumers remains unaffected. Raw water can be cooled for a practically unlimited period of time. The raw water make-up system, however, does not belong to the safety systems, so that under LOOP, it may not be available.

As it follows from the analysis of the ČSJ and the ESW failure, the EDU water systems, under the conservative approach (considering only halves of CČSI and CČSII, the level in the cooling towers on the minimum level), provide for a reserve sufficient for approximately 26 days for the production and make-up of demineralised water, and for approximately 39 days of residual heat removal (the ESW pumps operation) from the shutdown reactors without water make-up to the EDU systems. Therefore, the function of heat transfer to the ultimate heat sink is not subject to imminent risk.

In case it is impossible to overflow ESW in CHV to cool it down in the modes of the units´ after-cooling, heat will accumulate in the available water volumes (the ESW system and the CHV pools) as evaporation from the CHV pools surface is not sufficient to remove residual heat even from a single reactor. With the possibility to make up water to the ESW suction tanks, however, maintenance of acceptable temperature of ESW is possible for a period exceeding 72 hours.

Under LOOP, however, it will not be possible to count on raw water supply from ČS Jihlava. Without cool water make-up, heating of ESW will occur. At the conservative-assumed initial temperature of ESW 29 °C, the ESW heating-up to the temperature of 33 °C would occur (the calculated value based on conditions for the consumers operation) after the reactors´ scram in approximately 2 hours. With the use of the reserve from the decantation tanks, it is possible to acquire 3 more hours of operation of one ESW division. Increase of the TVD´s temperature would result in the TG´s overheating, which, however, can be compensated successfully by proportional reduction of load thereof. If the aforementioned is not accepted and if no other provisions are implemented (higher intensity of the room´s ventilation, portable air-conditioning units), the DG´s overheating might result in gradual shutdown thereof.

Loss of the TVD system

The ESW system is therefore of key importance from the point of view of provision of safety and residual heat transfer, either from the fuel located in the core, or from the fuel deposited in SFP to the ultimate heat sink.

With respect to the 3x100 % ESW systems redundancy and the additional redundancy of each ESW division (4 pumps), the loss of ability to transfer heat from the sources is conditioned by unavailability of all the ESW pumps (in total 12 pumps). 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 ESW pumps is highly unlikely. Even in case of operation of only one pump in one division of the ESW system, it is possible to cover the basic safety functions. The only possible cause of loss of all the ESW pumps could be SBO.

If irrecoverable leakage of ESW occurs, which cannot be compensated by switching to another division of the system, it is possible to count on maximum make-up of raw water from ČS Jihlava to reduce the speed of level decrease in the ESW tank in cases when the loss of power supply does not occur. Even provided that the pipeline from the ESW pumps to the consumers is not damaged, this intervention only reduces the speed of the tanks levels decrease and in 21 hours the whole water reserve at the ESW pumps suction in the given CČS (for one HVB) will have been exhausted. The speed of the level decrease in the tanks depends on the ESW leakage intensity. In case of the same event on all the units simultaneously, this period would be shorter.

5.1.2.2.1 Availability of an alternate heat sink

The total loss of ESW does not mean an imminent issue (see SBO as well) with respect to the possibility of a long-term residual heat removal to the atmosphere through SG after the units shutdown.

Among the main non-technological means applicable in case of loss of the ultimate heat sink, there is the pumping equipment of the On-Site Fire Rescue Brigade (FB). Apart from the possibility of emergency water make-up to SG through the emergency feed water system (for direct heat removal to the atmosphere) this equipment is not designed to provide for alternative methods of heat removal from the ESW consumers.

In the EDU location, there are no alternative or mobile sources to provide for circulation or heat removal from the ESW consumers, which could be used to manage the risks resulting from the loss of the ultimate heat sink (SBO, radioactive media leakage to the NPP environment, etc.).

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

With the closed reactor, it is possible to remove heat from the reactor on a long-term basis in the mode of heat removal through SG, including the possibility to cool the unit up to the temperature of approx. 110 ℃. With respect to the quantity of combinations to provide for this mode, diversified from the point of view of solution as well as redundant from the point of view of the systems configuration, direct risk is not identified for the provision of the heat removal safety function with the closed reactor.

Note: For heat removal with the closed reactor, it is possible to use alternatively the so-called Feed & Bleed (which means the SG supply with feed water by means of the FB equipment and steam removal through steam dump to atmosphere).

With the open reactor when heat removal from core is dependent on the ESW system operation (cooling in the PC mode, heat removal in the heat exchanger of the after-cooling system in the water-water mode on the secondary circuit), the consequence of the ESW loss is increase of temperature in core. In this case, it is possible to begin to fill up the refueling cavity (BVP) by means of the I.O emergency feed water systems from the ECCS (in total up to 1240 m3 of boric acid solution is available - depending on the condition of the technological systems upon refueling) and thus to postpone the temperature increase. Heat removal from the core can be maintained in this mode for a period exceeding 72 hours. If heat removal through TVD is not recovered, temperature in the SAOZ tanks and in BVP may rise up to the saturation level. In addition, safe condition of AZ is maintained by another very efficient strategy, namely by maintaining the level in the opened reactor by means of gravity-controlled coolant make-up from the bubbler tower´s conduits. The reserve of coolant to make up for the boiled away coolant is sufficient for approx. 12 days.

The loss of the ultimate heat sink as such, cannot endanger the containment integrity in case of modes with the closed reactor. The containment will start heating, however, its pressurization up to values when its integrity would be endangered (the design basis absolute pressure of 250 kPa) cannot occur. Cooling of the containment can be provided by means of the hermetic zone ventilation systems operation with coolers connected to the chilled water system.

In case of accident with leakage of coolant from I.O to the containment, its integrity is provided first of all by the spray pumps for the time they can suck from the ECCS tank. After switch to suck from the containment´s floor, the spray system efficiency will decrease with the increasing temperature of ESW. In case of failure of the spray system pumps, the passive containment spray systems by means of the vacuum bubbler system is available.

A different situation occurs in case of loss of the ultimate heat sink with the open reactor (during a refuelling outage) when no other barrier to absorb ionizing radiation and radionuclides is available. In such a case, there is a risk of possible leakage of radioactive media released from the BVP and SFP coolant during temperature maintenance on the saturation limit to the reactor hall or possibly outside NPP as well.

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

5.1.2.3.1 External actions foreseen to prevent fuel degradation

The independent means for media transport is the FB mobile equipment. It is possible to make up demineralised water directly to SG in an alternative manner using the fire water mobile pumps (with pressure at the pump delivery of 0.8 -1.2 MPa, flow rate120-150 t/h). Within the framework of the design completion, there are points of supply ready to interconnect the fire-fighting equipment with the technology. The alternative way of SG make-up is described in EOPs and has been tested in practice several times and the capacity of this equipment to secure basic safety functions has been verified.

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

See chapter 5.1.2.2.2

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

The ultimate heat sink in case of the EDU units 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 the sources of heat and the atmosphere is provided by the TVD system.

In the EDU´s location, the reserve of water sufficient for 39 days of the TVD system´s operation to remove residual heat from the EDU´s shutdown reactors without external water control of the TVD system. For one HVB (2 reactors), there are in total 12 TVD pumps available. A loss of all TVD pumps might result in a concurrent loss of power supply on both the units of the given HVB.

A loss of the ultimate heat sink occurs in case of a loss of the ability of the TVD system to transfer heat from AZ, BSVP and safety system equipment to the ambient atmosphere. For heat removal from AZ with the unit either in hot or semi-hot state in case of the TVD system´s loss, it is possible to use direct heat removal to the atmosphere through PGs, which are independent of heat removal by means of the TVD system. The loss of the TVD system is only associated with impossibility to cool down the unit to reach cold state, however, it is possible to maintain the unit in semi-hot state on a long-term basis.

With the open reactor when heat removal from core is dependent on the ESW system operation the consequence of the ESW loss is increase of temperature in core. In this case, it is possible to begin to fill up the refueling cavity (BVP) by means of the I.O emergency feed water systems from the ECCS (in total up to 1,240 m^3 of boric acid solution is available - depending on the condition of the technological systems upon refueling) and thus to postpone the temperature increase. Heat removal from the core can be maintained in this mode for a period exceeding 72 hours.

If heat removal through TVD is not recovered, temperature in the SAOZ tanks and in BVP may rise up to the saturation level. In addition, safe condition of AZ is maintained by another very efficient strategy, namely by maintaining the level in the opened reactor by means of gravity-controlled coolant make-up from the bubbler tower´s conduits. The reserve of coolant to make up for the boiled away coolant is sufficient for approx. 12 days.

The consequences of unsolved long-term loss of ability to remove heat to the ultimate heat sink might be at worst as follows:

Damage to fuel deposited in AZ and to spent fuel deposited in BSVP due to non-existence of alternative ways to remove heat from AZ, BSVP and components cooled by TVD (is there is no possibility to make up for boiled away coolant by means of the HZSp´s equipment).

The loss of cooling for the AC emergency power supply units (DGs) in case of LOOP may result in SBO (see chap. 5.1).

Leakage of radioactive media upon boiling from the opened reactor during outage or from BSVP to the surrounding environment.

Loss of ability to control the systems and the components communicating important parameters due to the loss of the SKŘ systems functionality in case it is impossible to remove heat losses from the SKŘ equipment.

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

The goal of the short-term measures is to eliminate the most critical risks by means of consolidation of the in-depth defence levels during initiating events exceeding the framework of the existing design (earthquake, floods, extreme conditions, human activity consequences, etc.), which may result in the loss of ability to fulfil the safety functions in case of loss of UHS.:

1. Propose and implement the diversified means to provide for cooling and heat removal from core and SFP, including the possibility of their connection to the existing technological systems.

2. Describe the use of the alternative and diversified means (proposed pursuant to paragraph. 1 – the so-called emergency plans (EDMG) with a view to providing for cooling and heat removal from core and from SFP.

The opportunities for improvement of the in-depth defence are provided in the following table. The table contains also areas for which it is necessary to execute supplementary analyses as at the time when the review was executed they were unavailable. A more detailed description of the corrective actions is provided in the annex.

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
Heat removal from AZ through II.O	Provide for sufficient source of the PG volume control	II	
Heat removal from AZ through I.O	Analyse the possibility of alternative make-up of the reactor using a pump and a new pipeline	II	
Diverse CHV	Implement provisions for diverse means of the ultimate heat sink (to CHV)	II	PSR finding
Regulations	Prepare the procedure for loss of UHS and the TVD systems on all the 4 units	I	
Regulations	Complete the existing procedures with the procedure for PG´s volume control for all the four units by means of fire-suppression equipment	I	
Regulations	In the existing procedures, specify the method of the gravity-controlled make-up of the opened reactor and BSVP from the XL conduits	I	Under implementation
Analyses	Heat removal from coolant in BSVP using coolant volume control and coolant accumulation in the TH tanks	I	Under implementation
Regulations	EDMG guidelines for the use of alternative means	II	

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

In case of a SBO event, the ESW pumps are not power supplied. As the ESW pumps are the intermediate element of the heat transfer to the atmosphere, then in case of SBO, the loss of the forced heat removal from I.O and SFP to the atmosphere will occur. A SBO event affecting the twin-unit will automatically mean the loss of the ultimate heat sink of the respective twin-unit due to the loss of power supply of the ESW pumps.

In case of the ultimate heat sink affecting the EDU twin-unit and the concurrent loss of power supply from the operational and the stand-by power supply units, the SBO situation will occur on the respective twin-unit due to the loss of the DG cooling system. The reason is the mutual dependence between DG and ESW - a failure of one of them will result in a failure of the other one.

For heat removal from the affected units, there is still the Feed & Bleed strategy on the side of II.O. This strategy is based on the possibility to supply SG by coolant using the FB means and to remove heat from I.O by means of the SG coolant evaporation and to remove the resulting steam through steam dump to the atmosphere. The ability to remove heat from SFP is completely lost with the exception of the possibility to make up for the evaporated water using the FB equipment. With respect to the fact that no other risks relating to the combination of the loss of the ultimate heat sink and SBO have been determined, the conclusions provided in the chapter covering SBO shall be applicable.

Loss of the ultimate heat sink due to SBO

The operability of the ESW system depends of the CHV integrity/functionality. The loss of the CHV function results in the limited ability to remove heat through ESW to the ultimate heat sink. Increase of the ESW temperature could lead to a gradual loss of all DGs. The only problem could occur in case of the concurrent LOOP situation when even SBO might develop gradually. The reason is the mutual dependence between DG and ESW - a failure of one of them will result in a failure of the other one.

Upon the SBO occurrence on one unit of the twin-unit, the loss of UHS may not occur as the ESW pumps of the other twin-unit will remain operable. Concurrent after-cooling of both the twin-units, for which the ESW system has been designed, is not assumed during SBO, so that the remaining two ESW pumps are able to provide for heat removal from the SBO-affected unit as far as their capacity is concerned. The possibilities to maintain the ESW throughflow to supply the consumers (the ECCS coolers, the SFP coolers, the technological condensers) of the affected unit are however difficult to use due to the failure of the pumps (the pumps of the normal or emergency after-cooling systems necessary to maintain the forced throughflow of media providing for heat removal to the ultimate heat sink on the side of I.O or II.O).

The selected consumers for heat removal both from I.O and from SFP (the SFP cooling system pumps or the after-cooling system pumps) can be alternatively power supplied from the other twin-unit (the method of power supply recovery is described in the applicable EOPs procedure) and therefore there is a real possibility that heat removal both from I.O and from SFP will remain unaffected and that the safety functions will be fulfilled on a long-term basis.

The I&C systems in accident states after the reactor scram are not all under operation so that waste heat generated by them would be lower. This reduces the need to cool the necessary I&C systems in a significant manner. The most important system PAMS has its own cooling system, which is power supplied from the category I power supply system as well.

An SBO event affecting both the twin-units always means the loss of all ESW pumps on the twin-unit, which means the loss of media removing residual heat from the coolers on I.O as well as II.O of the affected units to the atmosphere. However, there is still the strategy of heat removal from the core using the SG make-up by means of the FB equipment and the SG steam removal through steam dump to the atmosphere. In this mode, no heat removal on a long-term basis is provided for SFP. Without recovery of heat removal, the coolant in SFP would boil and uncovering of the fuel in the early stage of accident may be the consequence thereof (for detailed description, see SAMG). There is again possibility to keep level of boiling water in SFP using gravity make up from bubbler trays. Possibility to make up water in SFP is app. 13 days. The alternative approach is the use of fire-suppression equipment to make up for the lost coolant and to maintain the fuel temperature in SFP. EOPs contain this possibility of alternative SFP make-up, however, specific procedures for interventions on site have not been prepared yet.

SBO due to loss of the ultimate heat sink

The loss of the ultimate heat sink as such does not affect the power supply for the unit´s own consumption in case power supply from the operational or the stand-by power supply units is provided. However, if in case of the ongoing loss of the ultimate heat sink, the loss of the off-site power supply occurs and TG fail to get regulated to own consumption on neither of the twin-units, the emergency power supply units (DGs) are activated. For either twin-unit, there are three DGs available for the odd twin-unit and three DGs for the even twin-unit, which need the ESW throughflow for their operation. For the operation of DG, the lubrication oil temperature must be maintained (approx. 60 °C) and the internal circuit coolant temperature (83 °C).

In exceptional cases, which means during NPP emergency power supply, the grid breakdown, loss of power supply for the plant own consumption, etc. when it would be impossible to replace DG with another DG or another power supply unit, the DG could be operated with inactivated protection systems when only the protection system concerning the loss of the oil pressure would remain active. Without oil, the DG motor would seize up and would not be able to provide for power supply even after recovery of the ESW supply.

After the DG activation at 10th second within the framework of the ELS program, recovery of power supply for the two ESW pumps (of the respective unit and division) will occur. If the ESW throughflow fails to be recovered, DG could not be operated on a long-term basis. After connecting DG to the category II ZN switchgears and after load application, gradual heating of the coolant of the DG own circuit and lubrication oil would occur. In case of a gradual loss of the ultimate heat sink (e.g. the ESW heating-up due to impossibility to cool the heated water down by means of overflow), it is possible to maintain the temperatures by means of reasonable reduction of the DG load. In case of a sudden loss of ESW, DGs will overheat and lose their operability.

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

See chapter 5.1.1 and 5.1.2

5.1.3.2 External actions foreseen to prevent fuel degradation

Among other alternatives, there is the use of fire-suppression equipment to make up for the lost coolant and to maintain the fuel temperature in SFP. EOPs contain this possibility of alternative SFP make-up, however, specific procedures for interventions on site have not been prepared yet.

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 goal of the short-term measures is to eliminate the most critical risks by means of consolidation of the in-depth defence levels during initiating events exceeding the framework of the existing design (earthquake, floods, extreme conditions, human activity consequences, etc.), which may result in the loss of ability to fulfil the safety functions in case of SBO in combination with the loss of UHS.

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 5.1.3.6 and in case of the loss of UHS in chapter 5.2.5.

5.2 Spent fuel storage pools

The general description of the system is provided in chapters 1.3.1.2. and 1.3.3.

Residual heat from fuel assemblies deposited in BSVP is removed by the TG11, TG12 systems cooled by the 1st and the 3rd system of TVD.

5.2.1 Loss of electrical power

Upon total loss of power supply (SBO), a loss of the forced heat removal from BSVP through TVD to the atmosphere occurs. If SBO occurs on one unit only, it is possible to supply power to the pumps of the TG11, TG12 system from the twin unit through the service line. The procedure has been completed in EOPs.

From the point of view of provision for sufficient reserve of sub-criticality, SBO is not a problem. The geometry and material of the storage rack provide for sufficient sub-criticality even in case of boiling coolant or in case of filling BSVP with water with no content of H_3BO_3.

In case of loss of power supply, operability of the BSVP TG11, TG12 cooling systems is lost. Forced heat removal from SFP under SBO is interrupted immediately and a permanent increase of temperature, which is significant in particular in case of the upper grate fulfillment, occurs. Without recovery of heat removal, the coolant in SFP would boil and uncovering of the fuel in the early stage of accident may be the consequence thereof.

In case of failure of the SBO type, there is no design basis diverse system available. Heat removal is possible by means of alternative methods (see chap. 1.3.3.):

- upon a loss of the containment´s integrity, it is possible to provide for coolant supply by means of coolant bleeding from the hydroaccumulators
- make-up of BSVP from the higher-situated VBK conduits when it is possible to use the possibility of maintaining the BSVP level using gravity-controlled filling from the bubbler tower´s conduits. The reserve of coolant to make up for the boiled away coolant is sufficient for approx. 13 days. EOPs contain this possibility of alternative BSVP make-up, however, specific procedures for interventions on site have not been prepared yet.
- an alternative possibility is to use the fire-suppression equipment to make up for the lost coolant and to maintain the fuel temperature in BSVP. From this point of view, the pool is easily accessible for the HZS´s equipment (through the railway spur corridor). This extreme case of the method of BSVP´s cooling is based on the BSVP´s make-up with water supplied to the reactor hall using mobile pumping equipment with evaporation of the boiled away coolant back to the reactor hall

The emergency operating procedures EOPs contain the aforementioned methods of coolant make-up to BSVP, however, specific procedures for interventions on the spot have not been prepared yet.

The BSVPs are located in the reactor hall outside the containment in the reactor building. All fuel deposited in BSVP is with undamaged integrity, which is possible to prove in a demonstrable manner; in case of failed fuel, such fuel will be deposited in hermetically sealed containers in the BSVP´s rack forming a passive barrier against leakage of RA media and functioning passively as sufficient cooling of the deposited fuel assembly. Therefore, during performance of alternative activities to prevent damage to fuel by sufficient maintenance of level in BSVP by means of coolant make-up, possible boiling away of coolant from BSVP to the reactor hall will not result in significant release of RA media to the space of the reactor hall.

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

The goal of the measures is to eliminate the most critical risks by means of consolidation of the in-depth defence levels during initiating events exceeding the framework of the existing design (earthquake, floods, extreme conditions, human activity consequences, etc.), which may result in the loss of ability to fulfil the safety functions in case of SBO:

1. Propose and implement the alternative means of AC power supply for the existing equipment to provide for cooling and heat removal from core and SFP, including the posibility of thein connection to the existing power supply distribution system.
2. Propose and implement the diversified means for cooling and heat removal from core and SFP, including the possibility of their connection to the existing technological system.

The opportunities for improvement of the in-depth defence are provided in the following table. The table contains also areas for which it is necessary to execute supplementary analyses as at the time when the review was executed they were unavailable. A more detailed description of the corrective actions is provided in the annex.

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
Power supply of category I	Provide for sufficient power supply of the ZN systems of cat. I and the classified current consumers of ZN cat. II	II	
Regulations	Prepare the procedure for power supply recovery after SBO on all the units	I	
Regulations	Filling the open reactor and BSVP by gravity from the XL conduits	I	Under implementation

Regulations	EDMG guidelines for the use of alternative means	II	

The opportunities to improve the in-depth defence in case of SBO, which may result in a loss of ability to fulfil safety functions, are provided in the aforementioned table. The table contains also areas for which it is necessary to execute supplementary analyses as at the time when the review was executed they were unavailable.

5.2.2 Loss of ultimate heat sink

In case of failure of the TVD system, a loos of the forced heat removal from BSVP through TVD to the atmosphere occurs and it is not possible to remove heat from BSVP in a standard manner.

From the point of view of provision a sufficient reserve of sub-criticality, the loss of UHS is not a problem, see 5.2.1.

In case, none of the pump-heat exchanger-TVD combinations is able to remove residual heat from the fuel, it is possible to secure heat removal by means of coolant bleeding from BSVP to TH tanks of the SAOZ system with heating of water in these tanks. In doing so, no undesirable mixing of media occur as the TH tanks of NT are designed for the BSVP make-up in the refuelling modes (solution in these tanks serve for the purpose of increase of level in BSVP in the refuelling mode). For the purpose of coolant bleeding from BSVP, it is possible to the dedicated pump TG10D01. This procedure would be applied until the temperature in all SAOZ tanks has risen to 60 ℃. In this manner, it is possible to extend the time until the water in the pool starts boiling. In case of insufficient cooling of BSVP, the respective procedure included in EOPs has been prepared.

Another method considered for heat removal from the pool is the BSVP´s make-up from the VBK conduits, which can be provided using the 10D01 pump to supply the VBK XL conduits.

Another possible method how to make up BSVP is to use the TM13(14)D01TM pumps serving for the purpose of purification of the BSVP coolant and to supply coolant by means of these pumps from TH tanks of the NT SAOZ to BSVP. The TM pumps are power supplied from the systems sources of category II.

In case, the reactor´s integrity is lost during refuelling, there is also the possibility to supply coolant by means of an arbitrary pump of the VT or NT SAOZ system directly to the reactor interconnected with BSVP and from here to I.O, or it is possible to provide the coolant´s supply by means of bleeding from the hydroaccumulators.

If coolant from all the SAOZ tanks and bubbler conduits is used, the reserve of coolant is sufficient to make up for losses due to the coolant´s boiling away in BSVP for more than 8 days, even in case the fuel is arranged in two racks one above the other.

If none of the aforementioned methods is available, the cooling of BSVP is possible by alternative means, the same as in case of SBO:

- BSVP volume control from the higher-situated VBK conduits by gravity
- in case of reactor deprived of its integrity, it is possible to supply coolants by means of the hydroaccumulators´ bleeding
- BSVP volume control using the HZSp´s equipment.

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

The goal of the measures is to eliminate the most critical risks by means of consolidation of the in-depth defence levels during initiating events exceeding the framework of the existing design (earthquake, floods, extreme conditions, human activity consequences, etc.), which may result in the loss of ability to fulfil the safety functions in case of loss of UHS:

1. Propose and implement the diversified means to provide for cooling and heat removal from core and SFP, including the possibility of their connection to the existing technological systems.
2. Describe the use of the alternative and diversified means (proposed pursuant to paragraph. 1 – the so-called emergency plans (EDMG) with a view to providing for cooling and heat removal from core and from SFP.

The opportunities for improvement of the in-depth defence are provided in the following table. The table contains also areas for which it is necessary to execute supplementary analyses as at the time when the review was executed they were unavailable. A more detailed description of the corrective actions is provided in the annex.

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
Diverse CHV	Implement provisions for diverse means of the ultimate heat sink (to CHV)	II	PSR finding
Regulations	Prepare the procedure for loss of UHS and the TVD systems on all the 4 units	I	
Regulations	In the existing procedures, specify the method of the gravity-controlled make-up of the opened reactor and BSVP from the	I	Under implementation

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
	XL conduits		
Analyses	Heat removal from coolant in BSVP using coolant volume control and coolant accumulation in the TH tanks	I	Under implementation
Regulations	EDMG guidelines for the use of alternative means	II	

5.2.3 Loss of ultimate heat sink, combined with station black out

This case fully corresponds with the condition described in 5.2.1 as SBO is accompanied by a loss of the TVD system (UHS).

5.2.3.1 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

This case fully corresponds with the condition described in 5.2.1.1 as SBO is accompanied by a loss of the TVD system (UHS).

6 Severe accident management

For clear understanding of the following text, it is necessary to get familiarized with the content of chapter 1.3 describing the technological systems necessary for the provision of fulfilment of main as well as support safety functions of the Dukovany NPP.

6.1 Organization and arrangements of the licensee to manage accidents

6.1.1 Organisation of the licensee to manage the accident

The basic purpose of the NPP´s safety is to prevent uncontrolled leakages of radioactive media, in particular those which are generated in the AZ of the reactor. For the provision of this purpose, the design is based on the so-called "in-depth defence" concept utilizing the principle of multiple physical barriers preventing leakage of radioactive media. The purpose of severe accidents management is to provide the 4^{th} level of in-depth defence (mitigate consequences of a severe accident) following a failure of the 3^{rd} level of in-depth defence (which means failure of prevention of damage to fuel within the framework of management of design basis and beyond design basis events). The accident management system is followed by the emergency preparedness system, the main purpose of which is to provide the 5^{th} level of in-depth defence (mitigate radiation consequences of significant leakages of radioactive media).

Relationship between the physical barriers and the levels of in-depth defence:

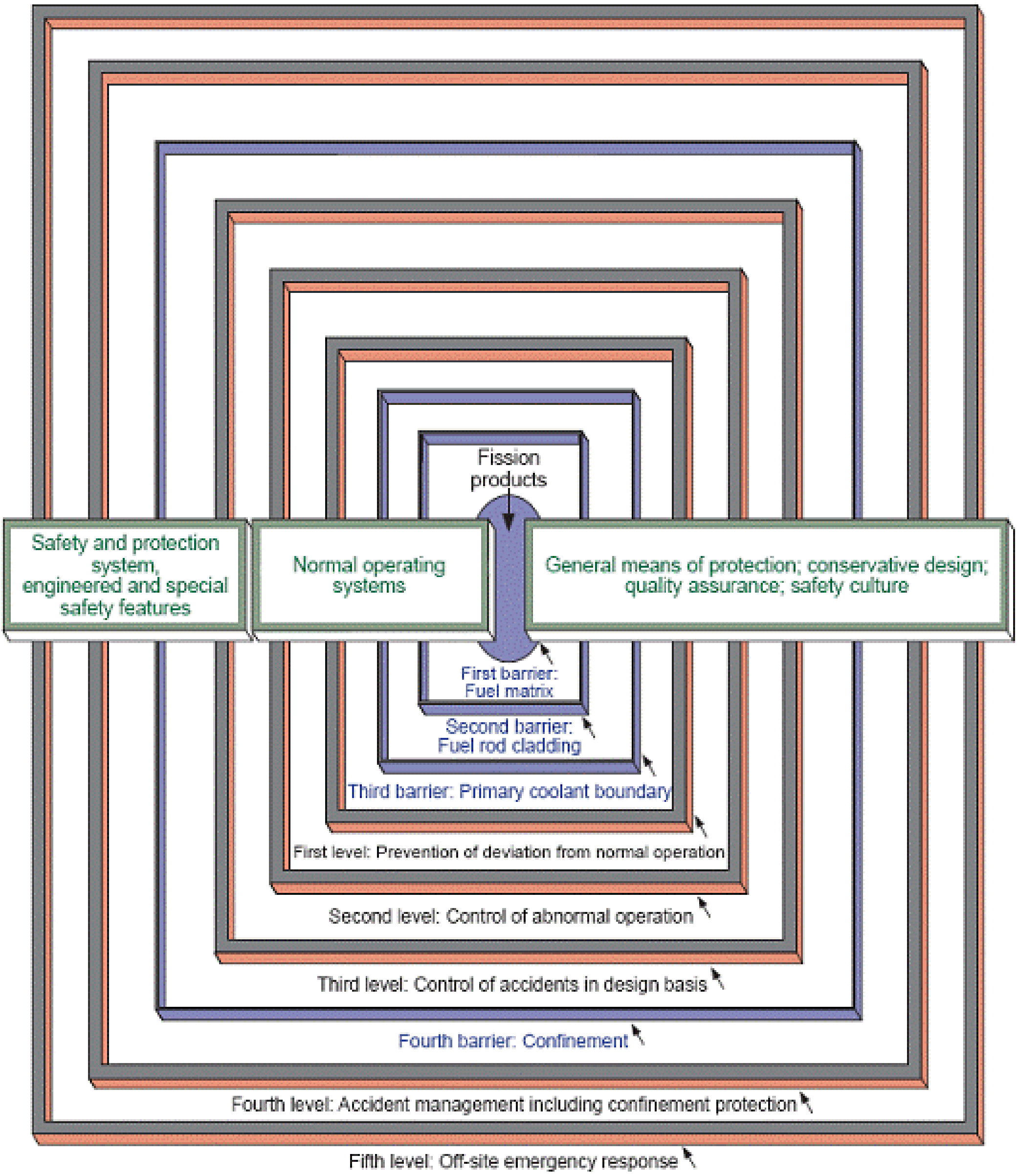


FIG. 4. The relation between physical barriers and levels of protection in defence in depth.

The EDU 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 centre (from which the staff provide for control and execution of interventions) and in the technical are, there is the provision of functional state of the required scope of technical means to implement the respective strategies.

6.1.1.1 Staffing and shift management in normal operation

The number of the staff members in the continuous shift operation of all the EDU units (shift staff) 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 so that the operating staffs have 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.

The operative management of the entire NPP is provided by the Shift Supervisor (SI) whose deputy in the shift is Safety Supervisor.

The EDU´s Shift Supervisor is responsible for execution of classification, emergency situation (MU) imposition and execution of activation of the necessary part of the EDU´s emergency response organization. If necessary, he is authorized to activate the respective part of the emergency response organization even sooner before the criteria for activation thereof are met. In the stage of MU´s development, SI may, based on supplementary specifying information, execute a change of classification depending on the actual situation.

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

- Reactor unit supervisor (VRB)
- Primary circuit operator (OPO)
- Secondary circuit operator (OSO)

The fundamental workplace of these personnel is the respective main control room. 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.

In case of MU imposition, the personnel of the EDU´s affected unit is complemented with a Safety Supervisor (BI), who shall assume responsibility for management of the technological equipment and become a contact person between the head of TPS and the main control room of the affected unit.

6.1.1.2 Plans for strengthening the site organization for accident management

The emergency planning system has been implemented in accordance with the IAEA requirements and methodologies and at the same time all legislative requirements applicable in the Czech Republic has been incorporated therein. Emergency planning counts among basic attributes of the nuclear power plants in the ČR. The purpose of HP in the NPP is to provide for technical, personal and documentary preparedness of the power station´s staff and of the external organizations participating in the solution of extraordinary events while accentuating the following:

- Prevention of serious damage to health due to extraordinary event
- Mitigating the risk of occurrence of extraordinary event or mitigating consequences of extraordinary event occurring in the NPP´s location and ZHP.

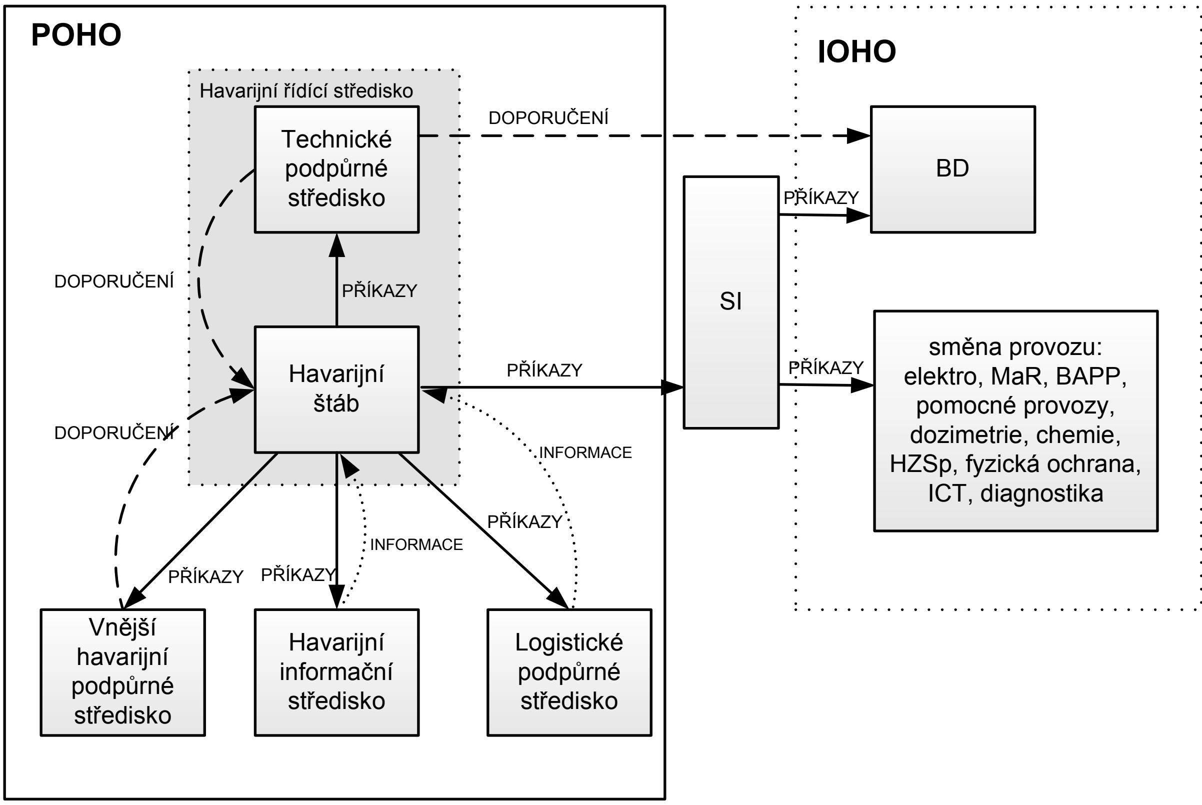
The strategic goals of the ČEZ company are transformed into the set long-term goals and tasks concerning the area of HP in accordance with the safety policy.

The strategy of HP is based on the logical development of any event occurring in an NPP. In case of occurrence of an extraordinary event, the respective intervention procedures have been prepared for the needs of management and execution of the intervention, or more precisely, the respective intervention instructions for employees, or other persons in the selected positions included in the Emergency Response Organization (OHO).

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

If the scope of the event exceeds the possibilities of the continuous shift personnel, IOHO is complemented with the on-call employees within the framework of the emergency response organization (POHO - standby emergency response organization). In this case, the following emergency response facilities are activated: Emergency Response Team, Technical Support Centre, Off-site Emergency Support Centre, Emergency Information Centre and the Logistical Emergency Support Centre. After activation of HŠ, the Commander of HŠ shall assume responsibility for the control of interventions.

Scheme of emergency response organization of the NPP, indicating mutual links and information flows

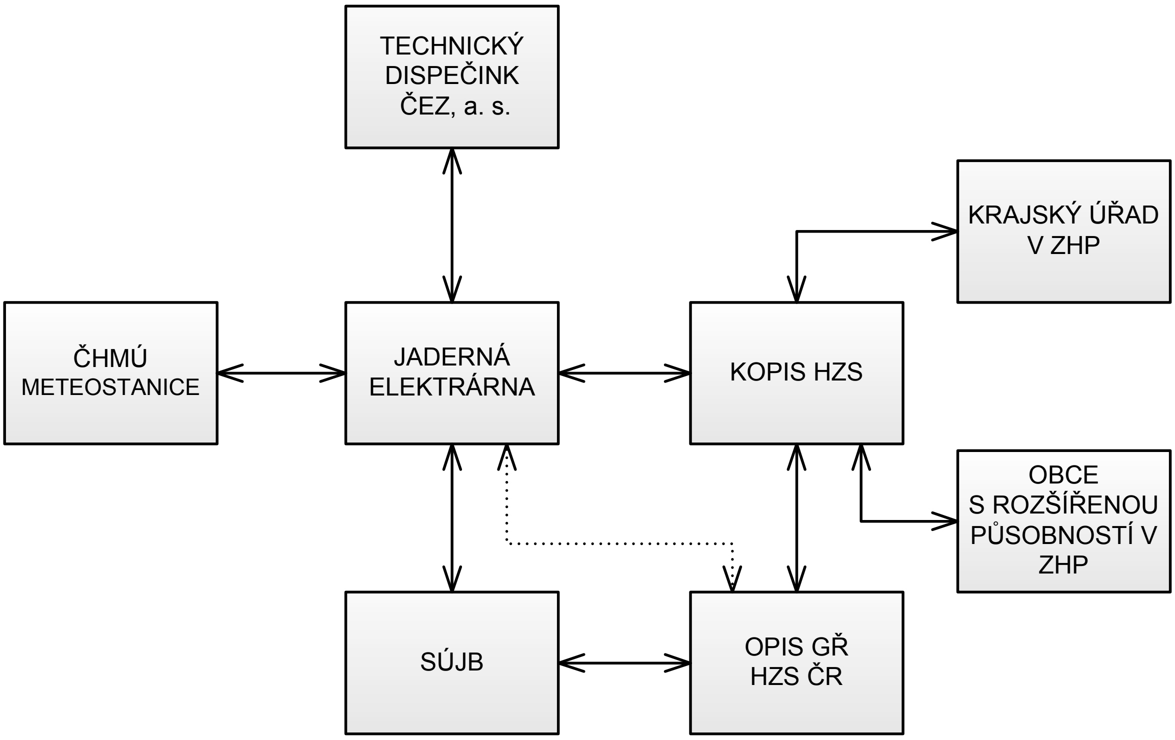


(Legend: Havarijní řídicí středisko = Emergency Control Centre, Technické podpůrné stredicko = Technical Support Centre, doporučení = recommendations, příkazy = commands, informace = information, Havarijní stab = Emergency Response Team, Vnější havarijní podpůrné středisko = Off-Site Emergency Support Centre, Havarijní Informační středisko = Emergency Information Centre, Logistické podpůrné středisko = Logistical Emergency Support Centre; box on the far right: shift of the operation: electro, MaR, BAPP, auxiliary operations, radiation protection, chemistry, HZSp, physical protection, ICT, diagnostics)

Upon occurrence of MU, SU shall inform the EDU´s management and ČEZ without any unnecessary delay; in addition he shall report the event to SÚJB, KÚ, the Regional Directorate of HZS, the respective municipalities with extended powers, TD ČEZ and the Meteorological station without any undue delay. The information shall be reported by means of the completed form for the Initial Report and the Follow-up Report on the extraordinary event. The forms are sent by e-mail or by fax. If direct communication with SÚJB is not possible, the standby path through the HZS´s operations centre (depicted in dashed line) is used.

A general view of the route of reporting to the authorities is shown in the figure.

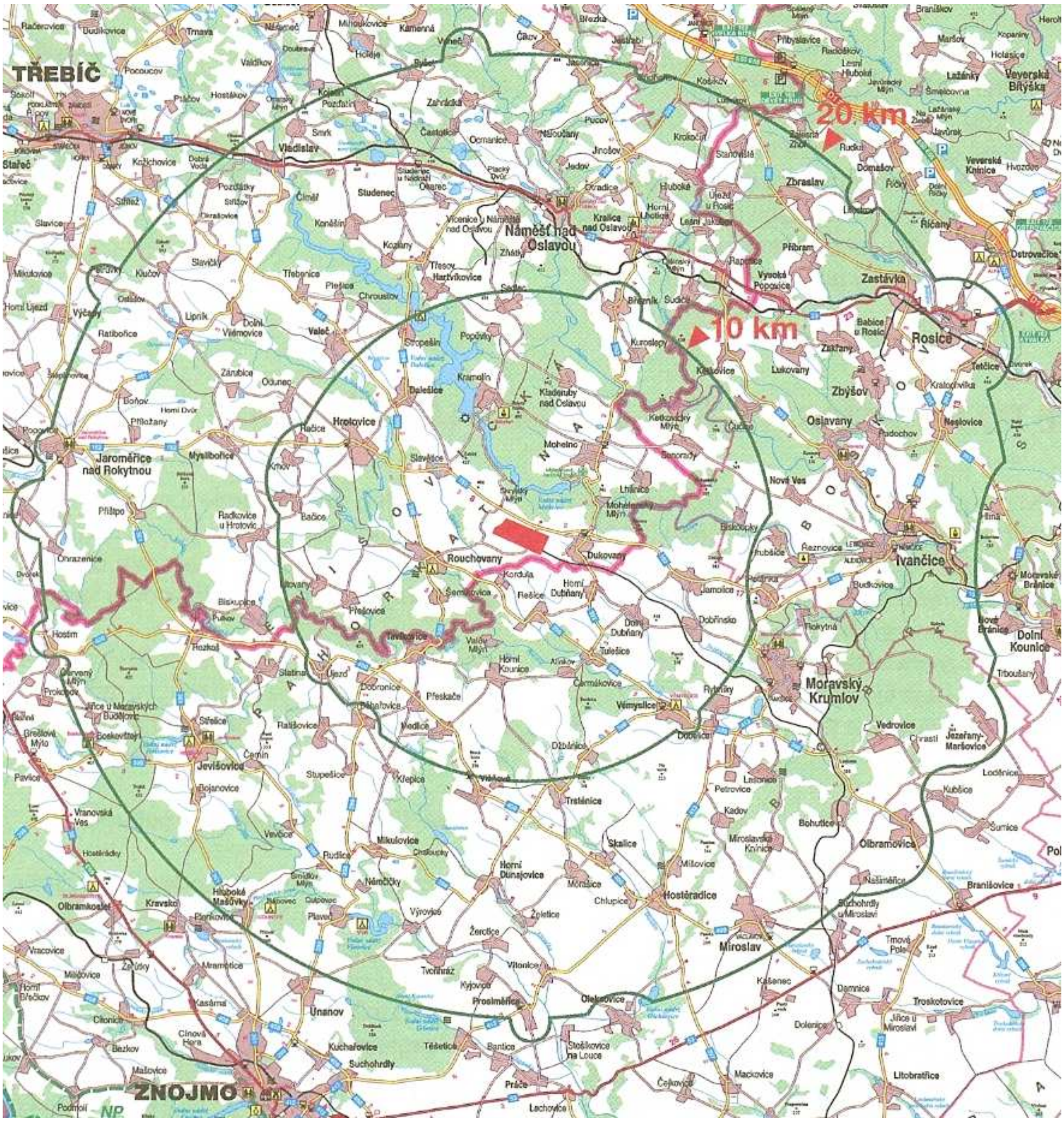
Representation of the route of reporting to the authorities upon occurrence of MU



(Legend: Technický dispečink ČEZ, a.s. = Technical dispatching ČEZ, a.s., ČHMÚ Meteostanice = Meteorological station of the Czech Hydrometeorological Institute, Jaderná elektrárna = nuclear power station, Krajský úřad = Regional Authority, Obce s rozšířenou působností = municipalities with extended powers, GŘ = Inspectorate General, ZHP = emergency planning zone)

For the needs of planning the provision of protection of population in the surrounding country of the NPP in case of radiation accident and for the need of preparation of the off-site emergency plan, the emergency planning zone of the NPP for EDU has been determined by decision of SÚJB as a territory within a radius of 20 km. Based on the same decision, to secure provisions for preparation and execution of evacuation of the population, the internal on-site part of ZHP given by the area delimited by a circle having a radius of 10 km, including the municipalities on its boundary was determined as well.

Emergency planning zone:



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 engineer 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 (OHO) 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 ECC in EDU 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 SÚJB 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 Commander of the Emergency Response Team 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.

Types of extraordinary events

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

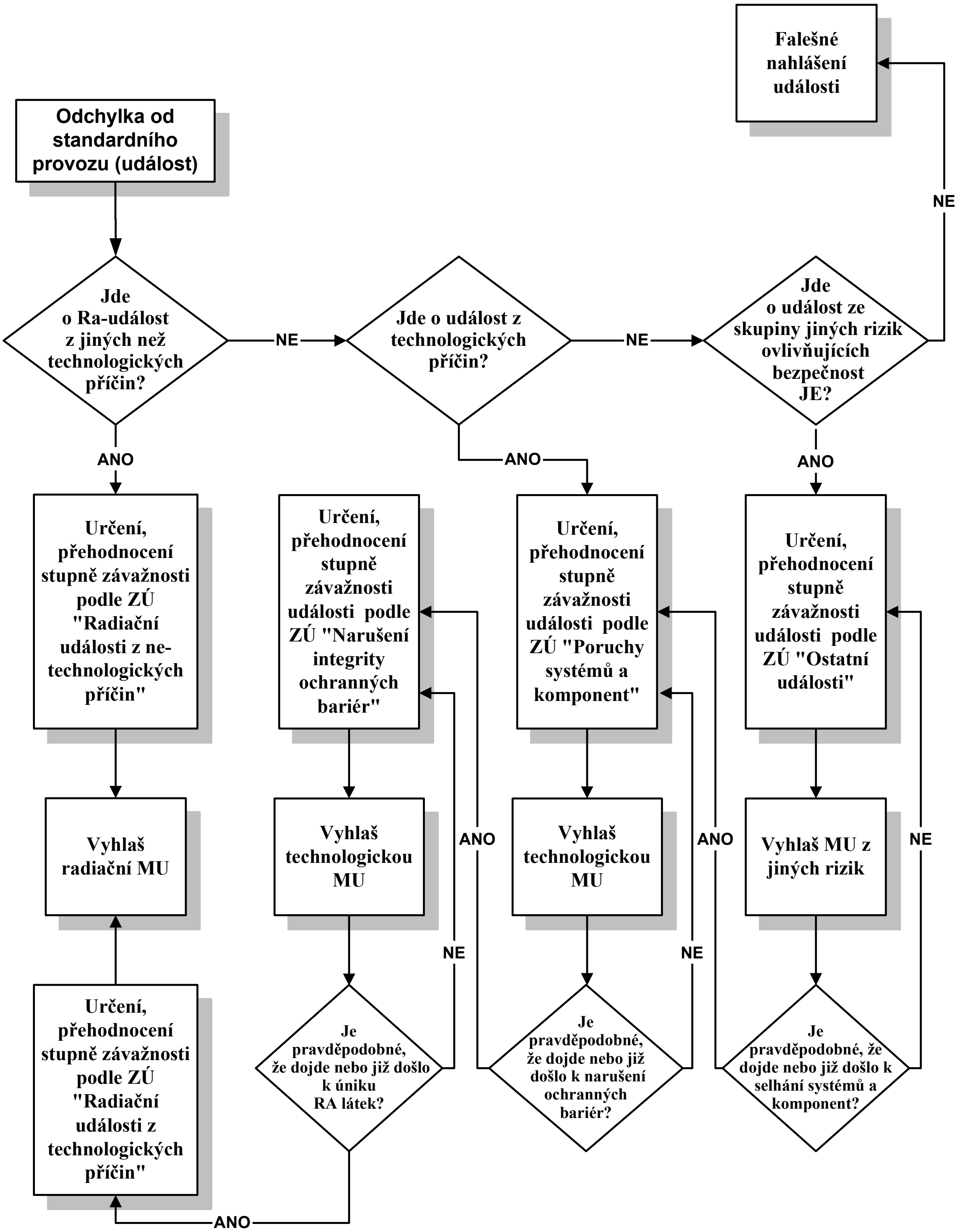
This classification of intervention levels allows the shift supervisor to identify the severity of the specific extraordinary event more easily, in particular from the point of view of provision of nuclear safety and radiation protection.

If the criteria of extraordinary event are met or in case of suspect occurrence thereof, the Shift Supervisor shall first verify whether a radiation-related extraordinary event due to non-technological causes is concerned. If this possibility is excluded, the Shift Supervisor shall verify whether an event due to technological causes is concerned which could result in damage to protective barriers and, if applicable, to any subsequent release of radioactive media, i.e. radiation event due to technological causes.

If an extraordinary event due to technological causes is confirmed (i.e. event associated with the nuclear fission process), the Shift Supervisor shall first check whether the intervention level “System and component failure” has occurred. When the applicable intervention level is identified, he shall declare the corresponding level of extraordinary event. At the same time, he shall check whether any of the intervention levels “Impairment of integrity of protective barriers” has occurred or could occur. If positive, he shall either confirm the declared level of extraordinary event or review it and declare a higher level. He shall proceed analogously in case of radiation-related intervention levels. Otherwise, i.e. if the achievement of the intervention level “Impairment of integrity of protective barriers” is not identified, he shall perform a periodic check of the correct identification of the intervention level “System and component failure”.

If the event is not a radiation-related extraordinary event due to non-technological causes or an extraordinary event due to technological causes, the shift supervisor shall check if the reported event corresponds with any of the pre-defined intervention levels in the category "Events due to other risks". When the applicable intervention level is identified, he shall declare the corresponding level of extraordinary event. At the same time, he shall check whether, in consequence of the detected MU due to other risks, there is a probability that some systems or components could fail or have already failed. The principles of further assessment are then identical with the process applicable in case of events due to technological causes.

Fundamental diagram of the Shift Supervisor's steps to assess the type of extraordinary event



(Legend: ano = yes, ne = no, by lines:
line 1: False report on event,
line 2: Deviation from standard operation (event)
line 3: Is a Ra event due to other than technological causes concerned?, Is an event due to technological causes concerned?, Is an event from the group of other risks affecting safety of the NPP concerned?
line 4: Determination, reassessment of the event´s level of severity according to ZÚ “Radiation events due to non-technological causes”., Determination, reassessment of the event´s level of severity according to ZÚ “Impairment of integrity of the protective barriers”., Determination, reassessment of the event´s level of severity according to

ZÚ “Failures of systems and components”., Determination, reassessment of the event´s level of severity according to ZÚ “Other events”.
line 5: Impose radiation MU, Impose technological MU, Impose technological MU, Impose MU due to other risks.
line 6: Determination, reassessment of the event´s level of severity according to ZÚ “Radiation events due to technological causes”, Is it likely that RA media leakage will occur or has occurred?, Is it likely that impairment of the protective barriers will occur or has occurred?, Is it likely that failure of systems and components will occur or has occurred?)

In case of imposition of level 1 MU, only the technical component of POHO (the TSC - Technical Support Centre) is activated; in case of imposition of Site Emergency or General Emergency, the remaining part - EDU Emergency Commission - is activated as well. Until its activation, activities are controlled by EDU shift engineer and the shift staff proceeds according to the respective operating procedures

SI shall be responsible for all activities relating to solution of MU until the HŠ Commander assumes the responsibility for the MU´s solution. SI shall remain responsible for the control of the technological equipment on the unaffected units. Accident mitigation of the affected unit is managed by BI. SI is responsible for the fulfilment of commands of the HŠ Commander in the area of management and coordination of the shift staff´s activities.

Until responsibility is assumed by the HŠ Commander, SI shall provide for the following:

- provides the prescribed information
- coordinates cooperation of HZSp, Medical Rescue Service, G4S Security Service
- in case of occurrence of MU 3. level 3, he shall provide for warning employees in the EDU´s premises as well as members of the public within ZHP
- if necessary, he shall manage the staff´s hiding in the shelters.

The workplace of TSC and EC is the Emergency Control Centre (ECC) located within EDU premises. With imposition of levels Site Emergency or General Emergency, the following entities are activated as well: the so-called Logistical Emergency 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 Support Centre (provision of radiation monitoring in EPZ) in Třebíč. All the aforementioned centres are controlled by the Emergency Commission.

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

To resolve technological accidents (up to fuel damage), strategies contained in the Emergency Operating Procedures (EOPs) have been developed. For extenuation 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 reactor 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.

<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 MU occurs, the shift supervisor shall be responsible for the imposition of MU and the management of activities in accordance with MU until his responsibility is handed over to the activated Commander of the Emergency Response Team. 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.

In case of imposition of extraordinary event, the continuous shift operation staff (with the exception of the supervising staff of the shift in BD), depending on the level of severity, either continues performing activities according the respective intervention instructions and instructions given by the supervising staff, or gathers in the shelter under the operational building, in case of imposition of safety measures, from where they perform the required interventions on the technological equipment or provide operational support to HZSp with respect to recovery and rescue works depending on instructions provided by SI or HŠ.

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, DGS operation.

<u>Standby emergency response organization</u>

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

- Emergency Response Team

The Emergency Response Team is the main managing body of the NPP's emergency response organization. After its activation, it secures imposition of protective measures for the employees and other parsons present in the NPP´s premises at the time of occurrence of the extraordinary event, management of activities of all employees and other persons participating in the intervention concerning suppression of development and mitigation of consequences of the extraordinary event in the nuclear power plant and provides for communication with the off-site services of the emergency response. The Emergency Response Team shall secure the supply of the necessary materials and special equipment, changing of personnel and provision of the necessary materials for them through the logistical support centre.

- Technical Support Centre

  The technical support centre is staffed so that it can provide qualified technical support to the personnel of the main control room of the affected unit to solve extraordinary events. 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 Response Team. The head of TPS may ask other specialists to support TPS through SI or the Commander of the Emergency Response Team.

- Off-site Emergency Support Centre

  VHPS provides for activities relating to radiation monitoring and assessment of radiation situation in the emergency planning zone and based on results of the radiation monitoring VHPS provides for forecasts of the radiation situation future development.

- Emergency Information Centre

  If an extraordinary event occurs, 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 in the premises of Atom hotel in Třebíč.

- Logistical Support Centre

  The personnel of the logistical support centre provides for the necessary technical means and qualified human resources based on the requirements and needs of the emergency response team, the technical support centre and the off-site emergency support centre.

The logistical support centre provides off-site support to OHO. The Logistical Support Centre is located in the premises of Atom hotel in Třebíč.

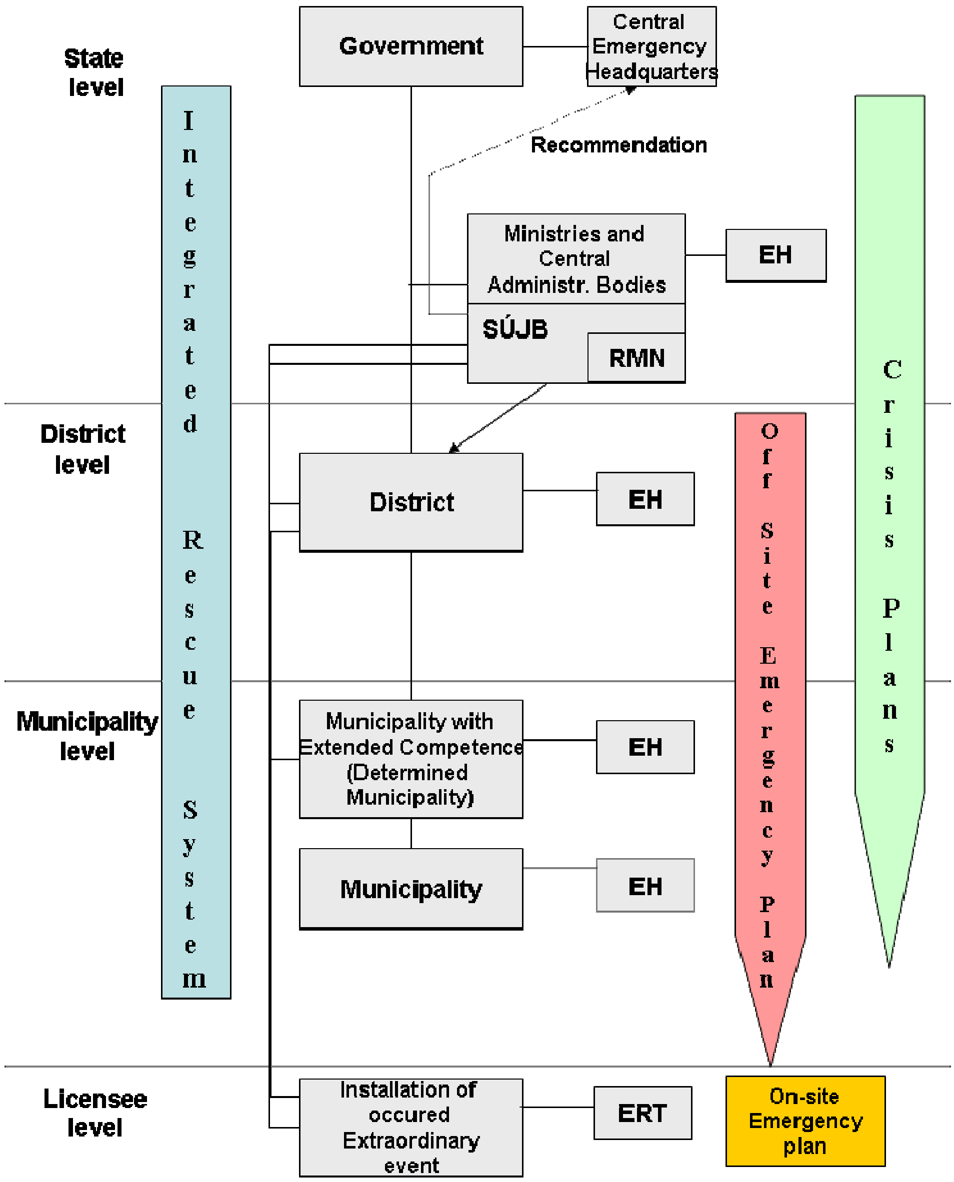
6.1.1.4 Use of off-site technical support for accident management

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 Emergency Support Centre.

Assistance in terms of transport or heavy equipment has been provided on the part of Regional Operations and Information Centre of “Vysočina Region” and “South Moravian 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, foreign assistance from other NPP of the VVER type located in the Slovak Republic - Bohunice, Mochovce, Hungury - Paks etc.) would be provided in case of necessity. The most efficient assistance is assumed from the Temelin NPP.

Many authorities and organizations at the national as well as local level are involved in the provision of off-site emergency preparedness of the NPP.

Provision of the off-site emergency preparedness of the NPP



If MU occurs and as far as its subsequent solution thereof is concerned, the nuclear power plant communicates with the following off-site authorities and organizations, both on the national and on the local level.

- SÚJB - Crisis Centre

  SÚJB 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 powers whose territory overlaps with the emergency planning zone (ZHP). In cooperation with the mayors of the affected municipalities with extended powers, 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 SÚJB 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 powers

  The mayors of municipalities with extended powers shall decide on summoning the municipal crisis centres 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 powers, SÚJB 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 powers.

- Czech Hydrometeorological Institute

  The Czech Hydrometeorological 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.

- IZS - Integrated Rescue System

  The Integrated Rescue System (hereinafter referred to "IZS") has been built for the purpose of coordinated management and settlement of extraordinary events without detailed specification whether industrial accident, flood, earthquake or other natural disaster is concerned. From the point of view of legislation, the issue is covered by law regulating the integrated rescue system and the crisis management. Within the framework of IZS, the Central Emergency Plan of IZS has been prepared which is to be applied if, as a consequence of extraordinary event or a crisis situation or security action, the respective need occurs and if the statutory conditions for central coordination of rescue and mitigation works are met, or more precisely, if the president of the region, the mayor of the respective municipality with extended powers, the Regional Director of HZS or the rescue master ask, through the operations centre and the information centre of IZS, for help and forces and means unavailable to IZS on the level of the region to perform rescue and mitigation works during extraordinary event which is being settled independently in the respective region.

The forces and means upon centralized coordination of rescue and mitigation works are called in and deployed by the Ministry of the Interior - the Directorate General of the Fire Rescue Service of the Czech Republic (hereinafter referred to as the “Directorate General”) through its operations and information centre.

- Fire Rescue Service
  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 on ČT and ČRo. The regional fire rescue service shall also make sure for ČEZ, a.s. that the concerned municipalities with extended powers are informed through regional operation and information centres of the fire rescue service (in compliance with Decree No. 318/2002 Coll., as amended).

- Police, Security Information Service and Armed Forces of the Czech Republic
  Within the framework of IZS there are, among others, 6 helicopters intended for rescue works (Armed Forces of the ČR and the Police of the ČR) with the possibility to transport persons and load when 4 crews are on call with the possibility of activation within 10 minutes during the day and 20 minutes at night.

- Medical Rescue Service (Traumatological plan)

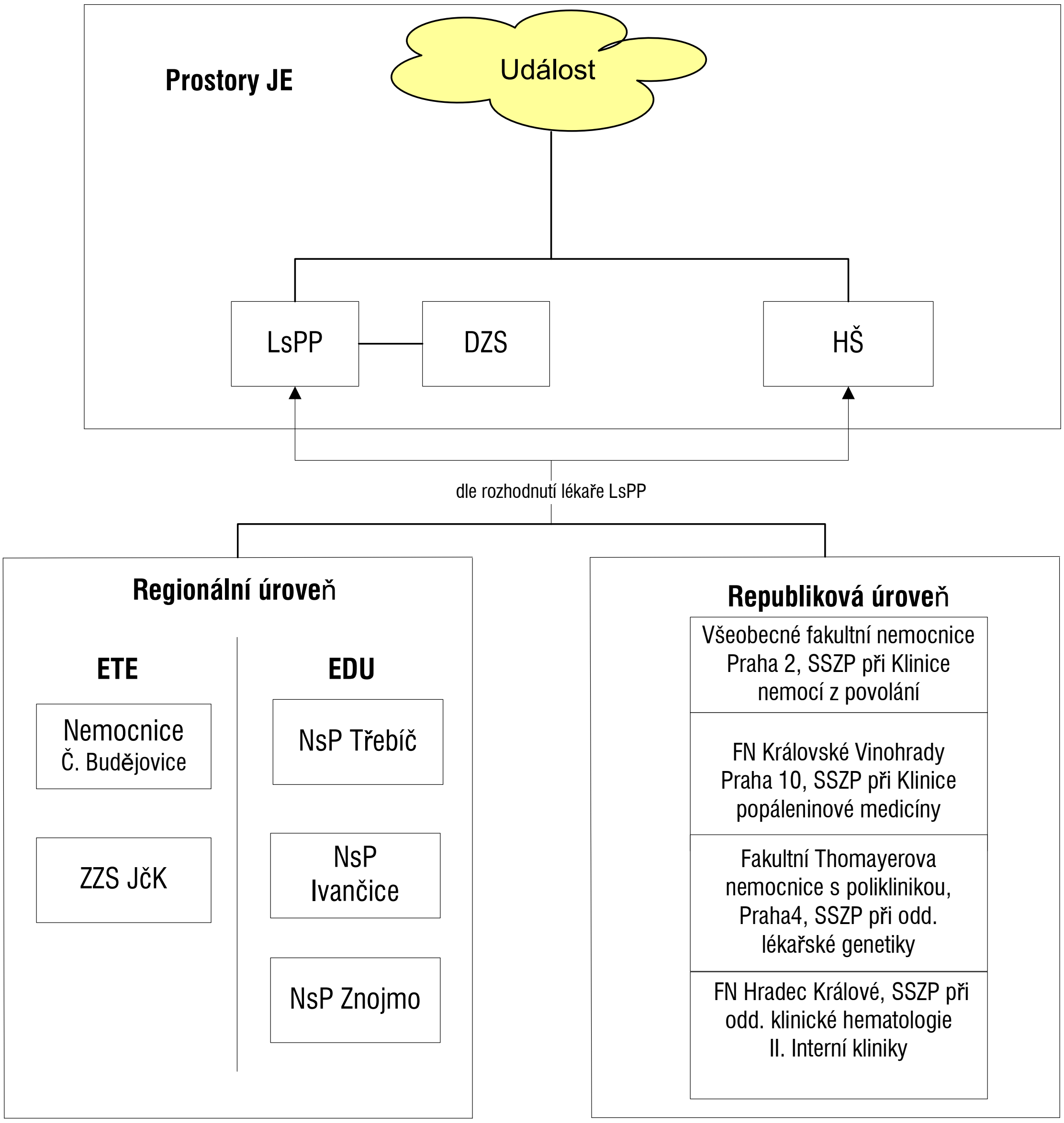
In the premises of EDU, the medical rescue service (hereinafter referred to as “LsPP”) has been established on a contractual basis with continuous emergency service, which is responsible for execution of medical service. During working hours, medical service is provided by resident doctors (2) and nurses (2), outside working hours, the service is provided by the contract physicians, or nurses. The LsPP is staffed on a continuous basis by the medical rescue service physician, nurse and ambulance driver. Upon occurrence of extraordinary event, the LsPP physician is responsible, among others, for the following:

- management and coordination of the medical aspect of the intervention,
- method of provision of first aid to employees and other persons affected by an extraordinary event,

- method of provision of medical treatment to employees and other persons affected by an extraordinary event,
- method of provision of specialized medical assistance to employees and other persons affected by an extraordinary event,
- cooperation with the intervening persons that provide for intervention management and performance as well as monitoring of the radiation situation.

The document also contains links and flows of information concerning solution of traumatological event in case of occurrence of extraordinary event and links to specialized medical centres (four "Centres of Specialized Medical Care" have been established).

Figure: Links to medical units and off-site medical facilities



(Legend: prostory JE = NPP premises, událost = event, dle rozhodnutí lékaře = based on the physician´s decision, regionální úroveň = regional level, nemocnice = hospital, republiková úroveň = national level)

In addition to professional medical service, for every HZSp EDU´s shift, there are 4 members of the shift trained to be able to provide first aid. Their activation shall be performed by the on-site fire rescue service's dispatcher, upon an order from the shift supervisor. When performing their working duties, having arrived at the site of a traumatological event, the above-mentioned members of the on-site fire rescue brigade shift shall, if needed, provide first aid to the affected person and, when a doctor arrives, cooperate with him/her and obey his/her instructions.

6.1.1.5 Procedures, training and exercises

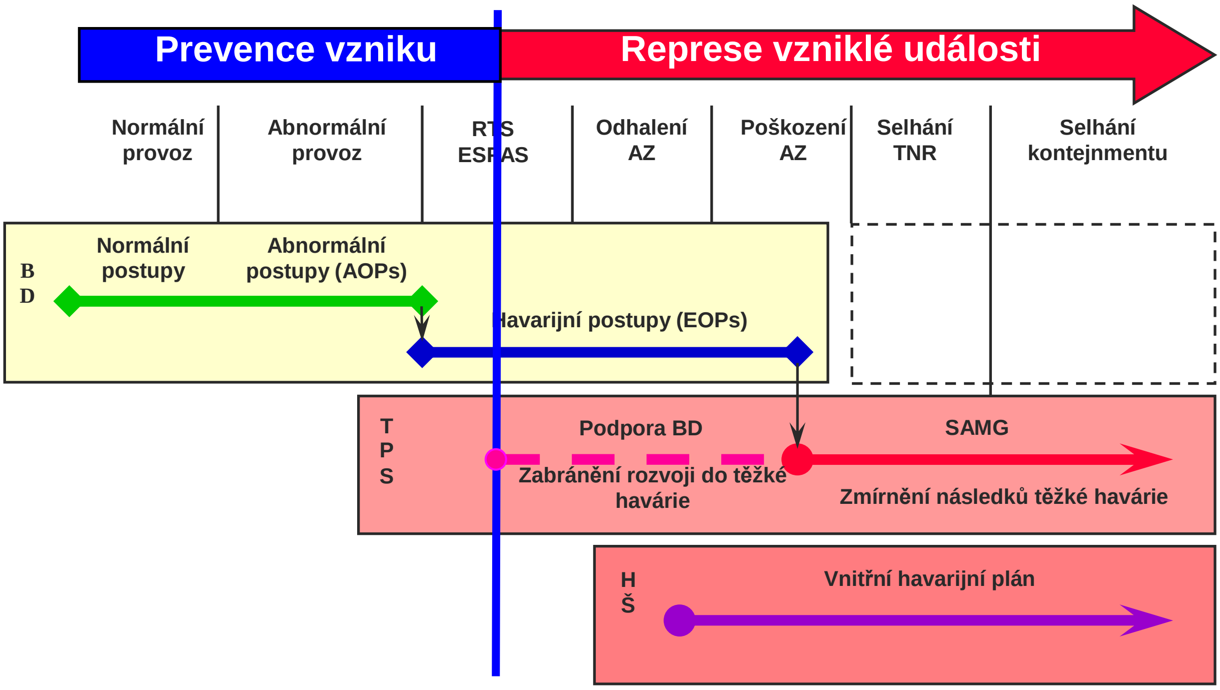
Accident management programme is based on symptom oriented approach. At the present time, the following strategies to manage design basis, beyond design basis and severe accidents are prepared for EDU.

- Symptom based Emergency Operation Procedures for power states (EOPs).
- Symptom based Emergency Operation Procedures for shutdown states, including cases of threat to heat removal from spent fuel deposited in SFP (SDEOPs).
- Manuals for decision-taking by TSC.
- Severe Accidents Management Guidelines for power states (SAMG).

All the aforementioned procedures and guidelines have been prepared and are updated in cooperation with the Westinghouse company.

In addition to the type of use of the operating documentation during activities taken in response to the respective situation, the procedure concerning emergency situation development is also closely related to activities of the emergency response organization based on the On-site emergency plan (imposition of a degree of the extraordinary event).

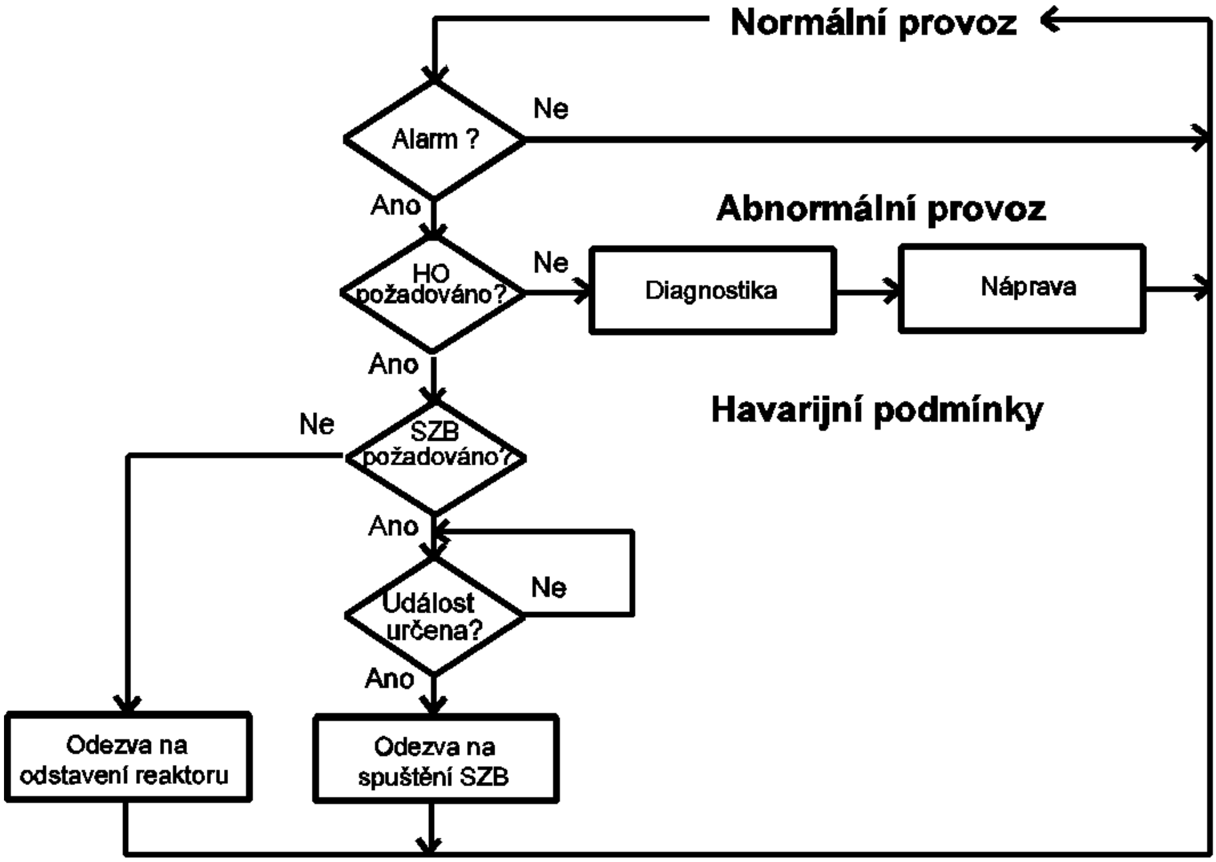
The link between the condition of the unit, the applied operating documentation and MU



(Legend: prevence vzniku = prevention of occurrence, represe vzniklé události = suppression of the occurred event, normální provoz = normal operation, abnormální provoz = abnormal operation, odhalení AZ = AZ exposure, poškození AZ = AZ damage, selhání TNR = TNR failure, selhání kontejnmentu = containment failure, normální postupy = normal procedures, abnormální postupy = abnormal procedures (AOPs), havarijní postupy = emergency procedures (EOPs), podpora BD = BD support, zabránění rozvoje do těžké havárie = prevention of development of a severe accident, zmírnění následků těžké havárie = mitigation of consequences of a severe accident, vnitří havarijní plan = on-site emergency plan)

All conditions of the unit in the prevention stage have common levels of response on the part of the operative personnel. Activities of the operative personnel at each level are controlled by the operating procedures accommodated to each operating condition. The procedures comprise a network determining activities of the operative personnel in each operating condition of the unit.

Method of response of the operative personnel



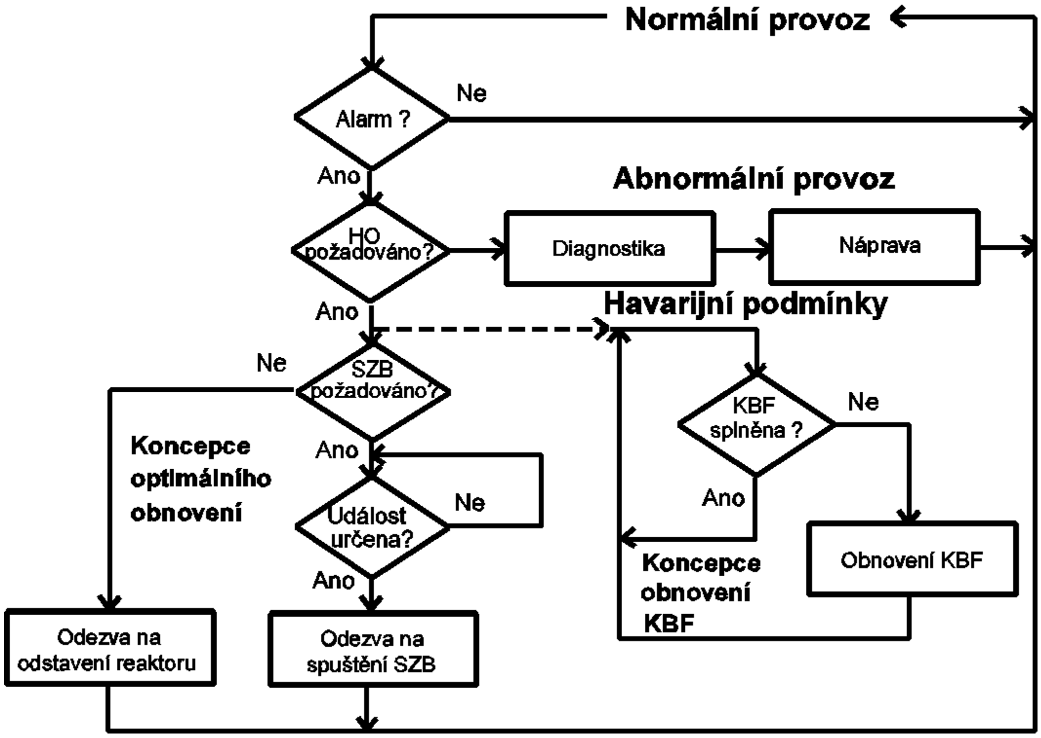
(Legend: normální provoz = normal operation, abnormální provoz = abnormal operation, havarijní podmínky = emergency conditions, diagnostika = diagnostics, náprava = remedy, ano = yes, ne = no, alarm? = alarm?, HO požadováno? = HO required?, SZB požadováno? = SZB required?, Událost určena? = Event identified?, Odezva na spuštění SZB = Response to SZB activation, Odezva na odstavení reaktoru = Response to the reactor shutdown)

For emergency conditions in the preventive stage, strategies contained in the Emergency Operating procedures (EOPs) have been developed. For management of severe accidents, the respective strategies have been developed, which are contained in the severe accident management guidelines (SAMG). The basic condition for execution of activities according to the emergency procedures is such a condition of AZ, which enables its cooling, which means that AZ is in a coolable geometrical configuration. If AZ is damaged in an irreversible manner, the emergency procedures may not provide for optimal guideline for the settlement of the emergency situation, and it is necessary to start activities imposed by SAMG. At this moment the main priorities change. In EOPs, the principal priority shall always be the recovery of heat removal from AZ 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 interventions described within the framework of EOPs, which the operative personnel will use to solve design basis as well as beyond design basis accident events, is to provide for sufficient cooling of AZ and thus to prevent irreversible damage thereto and in addition, to minimize consequences of possible leakage of radioactive media to the NPP´s surrounding environment. The philosophy of these procedures also contains permanent assessment of the condition of the physical barriers against activity leakage by means of assessment of the critical safety functions. This assessment provides for timely identification of deterioration of the safety condition of the unit and guarantees a possibility to perform early remedies if a negative trend of the event´s development is discovered.

The set of symptomatic-oriented emergency operating procedures provides the operative personnel with a systematic means (independent of the course of the emergency mode) to solve emergency situations. The combination of event-oriented and function-oriented strategies provide the operative personnel with a guideline to lead the unit to safe and final condition while providing for permanent diagnostics of the unit´s condition and possible recovery of the safe condition independently of the course of the respective accident event.

The concept of the unit control under emergency conditions

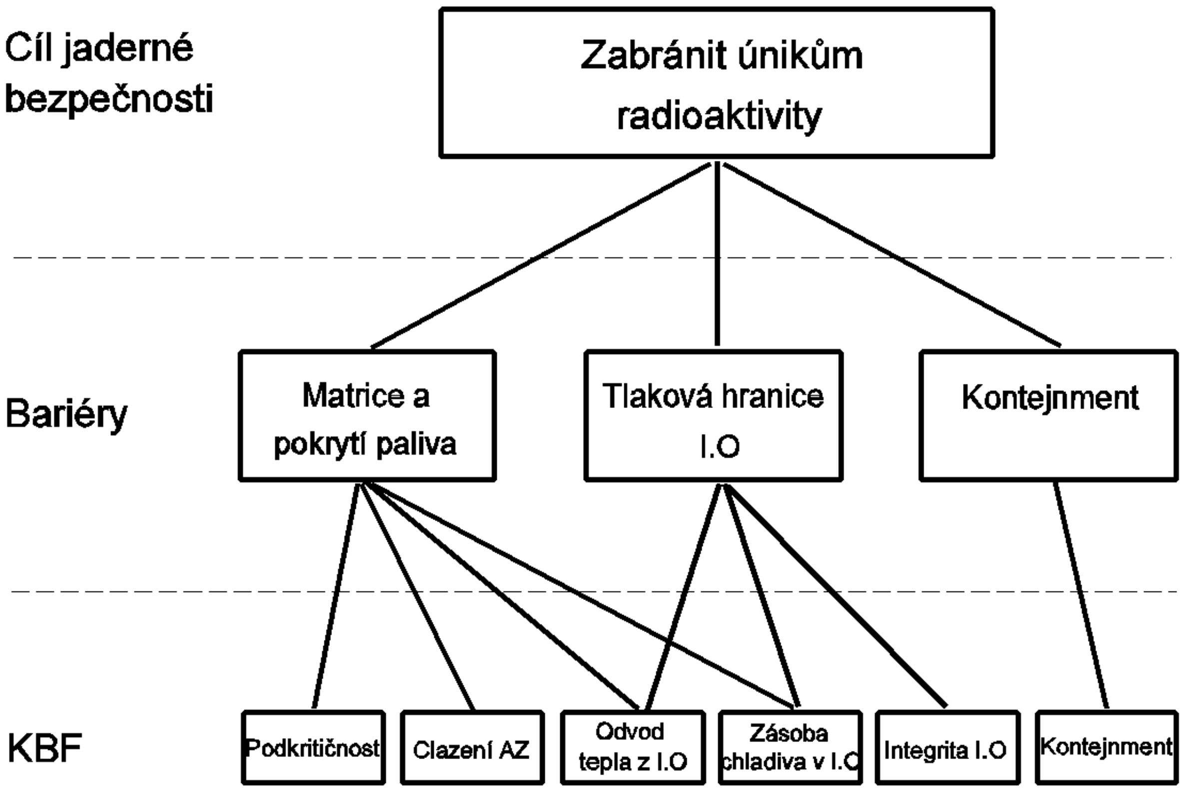


(Legend: normální provoz = normal operation, abnormální provoz = abnormal operation, havarijní podmínky = emergency conditions, diagnostika = diagnostics, náprava = remedy, KBF splněna? = KBF fulfilled?, Koncepce obnovení KBF = Concept of KBF´s recovery, Obnovení KBF = KBF´s recovery, ano = yes, ne = no, alarm? = alarm?, HO požadováno? = HO required?, SZB požadováno? = SZB required?, Událost určena? = Event identified?, Odezva na spuštění SZB = Response to SZB activation, Odezva na odstavení reaktoru = Response to the reactor shutdown, Koncepce optimálního obnovení = Optimal recovery concept)

The emergency procedures also contain a systematic means to assess the safe condition of the unit using assessment of conditions of the critical safety functions. The critical safety functions are closely related to the physical barriers that prevent leakage of radioactivity to the surrounding environment. If undamaged state of the matrix and the fuel cladding, the interface between the primary circuit and the containment is secured, then the power station does not put health and safety of the population at risk. However, if one or more barriers are violated, then the risk of exposure of the population increases. If integrity of all the barriers is lost, direct exposure of the population would be imminent and it would be necessary to perform extraordinary off-site emergency interventions. For these reasons, the purposes of the nuclear power station´s operation (in the meaning of nuclear safety) is to provide, to maximum extent and under any circumstances or events, undamaged condition of the physical barriers preventing leakage of radionuclides.

For each of the aforementioned physical barriers there is a set of safety functions that must be fulfilled to provide for undamaged conditions thereof. If the set of the safety functions is fulfilled, complete protection of the population against possible consequences of the nuclear power station´s operation is guaranteed.

Relationship between the safety functions



(Legend: line 1: Nuclear safety goal, Prevent radioactive media leakage
line 2: Barriers, Matrix and fuel cladding, Pressure limit of I.O, Containment
line 3: KBF, Sub-criticality, AZ cooling, Heat removal from I.O, Coolant reserve in I.O, I.O integrity, Containment)

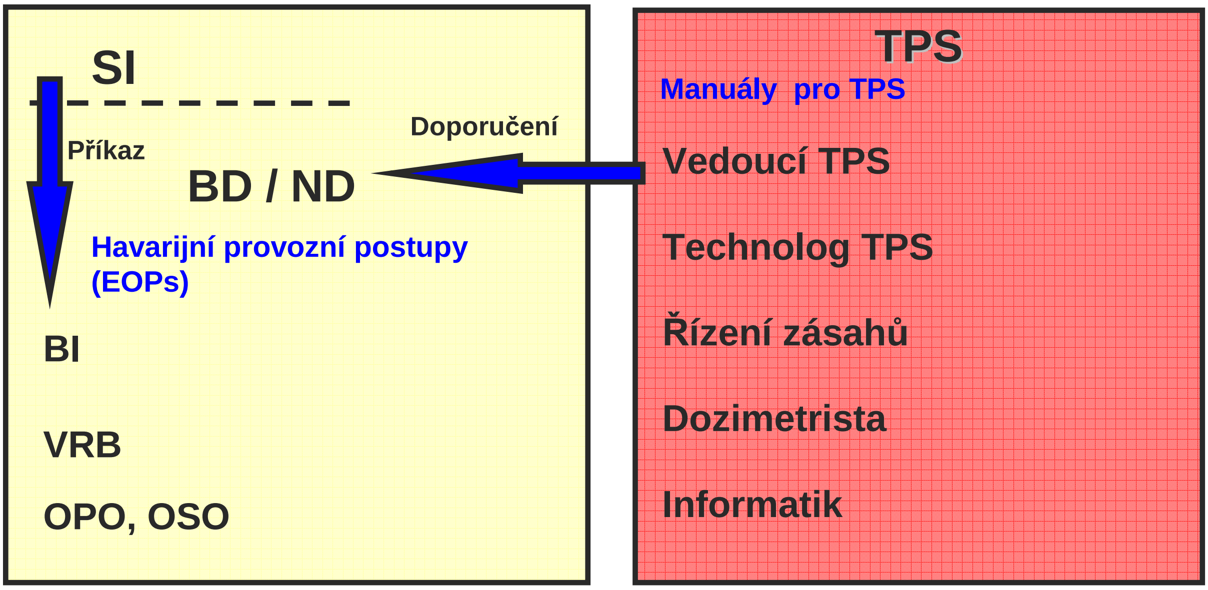
For the preventive stage of accident situation management, if the operating personnel proceeds according to EOPs, the TPS personnel has the guidelines at its disposal ("Manuals for TPS") which provide source material for decision-taking in support of the operating personnel with respect to execution of activities according to the emergency procedures. The emergency procedures contain numerous steps where instructions for further activities are specifically required from the TPS personnel. The experience resulting from training, simulations on the full scope operation training simulator, etc. also show that support from the TPS personnel is required in a number of other situations without a specific requirement in the respective step. In all the aforementioned cases, the decision depends on the current development of the emergency situation and on the specific condition of the systems and equipment of the unit concerned.

These guidelines have been prepared for the TPS personnel and for the other technical personnel of the NPP, who are, besides the TPS personnel, authorized to provide support for decisions. Among such personnel, there are BI, SI or VRB, who will find in this document a number of source materials for qualified decisions concerning other activities according to EOPs.

- The guidelines for TPS are used by the TPS personnel when MU1 is imposed, the TPS has been summoned and the TPS personnel is able to provide support
- The guidelines in this PP are used by BI, SI or by VRB if the support for decision is required before TPS becomes functional.

The TPS personnel provide support by assessing the current state according to guidelines and by handing over recommendations concerning the use of the emergency procedures.

Scheme of communication between TPS and the operating personnel for the use of the Manuals for TPS



(Legend: Příkaz = Command, Doporučení = Recommendation, Havarijní provozní postupy = Emergency operating procedures, Manuály pro TPS = Manuals for TPS, Vedoucí TPS = Head of TPS, Technolog TPS = Technologist of TPS, Řízení zásahů = Interventions management, Dosimetrista = Head of radiation protection, Informatik = Head of information technologies)

In case of development of events to become severe accident, another procedure is selected with respect to secure at least the remaining barriers against leakage of radioactivity. The loss of integrity and the geometry of fuel in AZ means a serious risk for the ability to remove heat from AZ. Under such conditions, it is not possible to continue proceeding according to EOPs. For such a stage of accident, SAMG have been prepared, by means of which activities to achieve a controlled stable condition are executed.

Transfer to SAMG is implemented in case irreversible damage to AZ has been discovered. In such a case, activities according to EOPs are terminated and transfer to SAMG is implemented. The only access point to SAMG is the SACRG-1 guideline, ACTIVITY OF BD WITHOUT TPS.

There are three possible transfers from EOPs to SAMG:

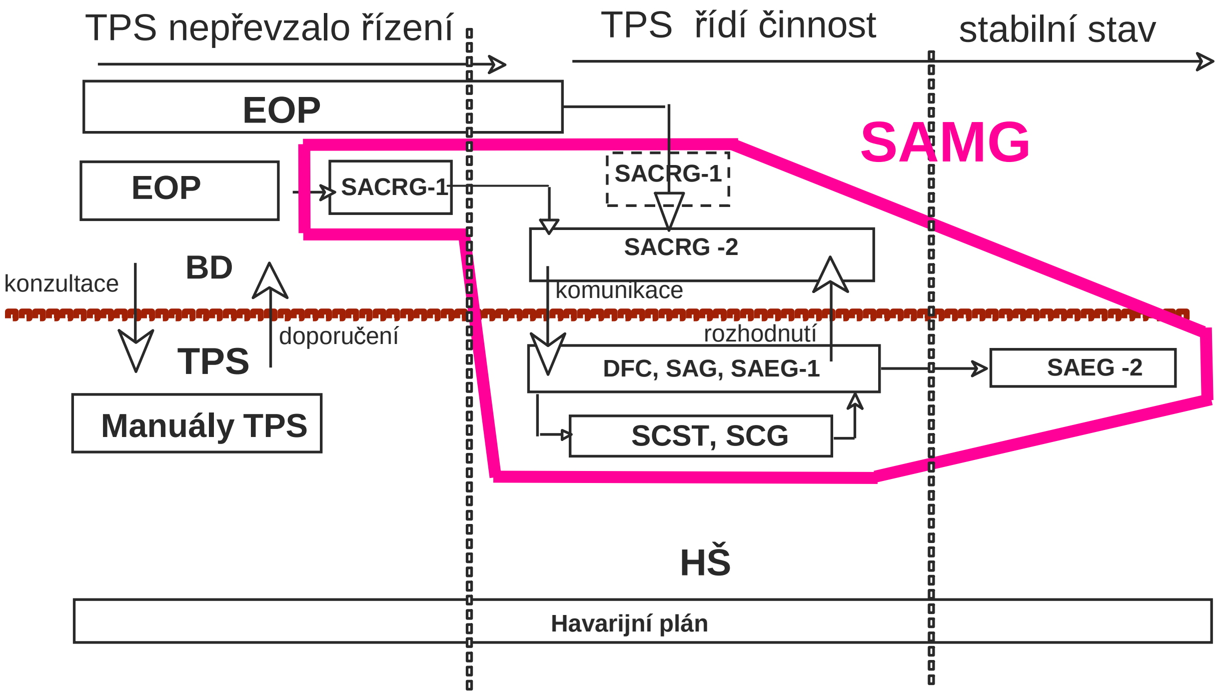
FR-C.1 Loss of cooling of AZ

FR-S.1 Failure to shut down the reactor (ATWS)

ECA-0.0 Loss of power supply - Blackout

These three possible transfers from the emergency procedures to SAMG are sufficient and cover all possible scenarios of severe accidents.

Links between EOPs and SAMG



(Legend: TPS nepřevzalo řízení = TPS failed to assume control, TPS řídí činnost = TPS is in charge of operations, stabilní stav = stable state, konzultace = consultations, doporučení = recommendations, Manuály TPS = Manuals for TPS, komunikace = communication, rozhodnutí = decision, Havarijní plan = Emergency plan)

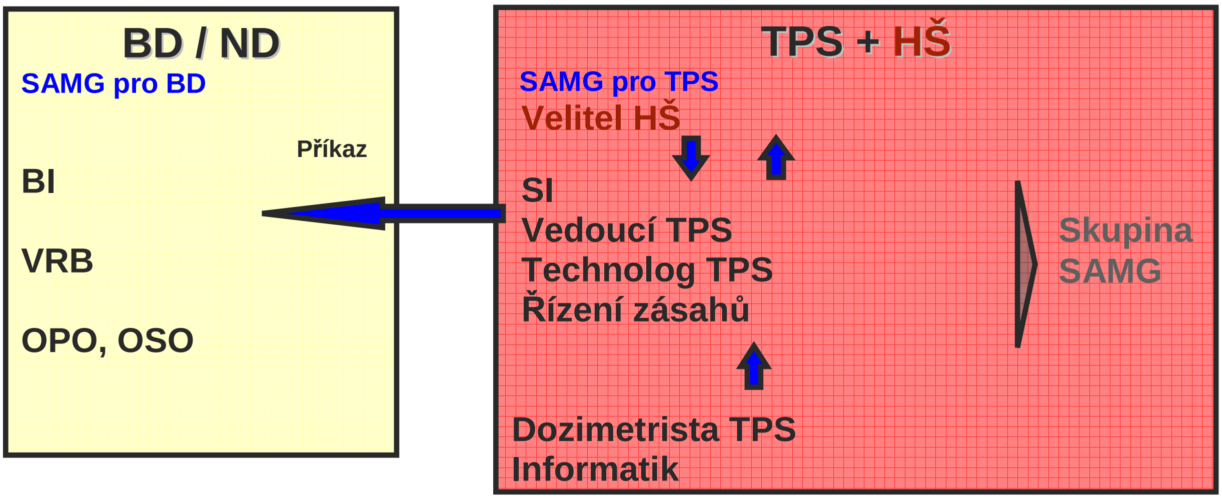
For mitigation of consequences of severe accidents, the following goals have to be fulfilled:

- Primary goals of SAMG
  - Recover removal of heat from AZ or melted material thereof = return the source of heat to the controlled and stable condition
  - Maintain the integrity of the containment as the last barrier against leakage of Ra media to the surrounding environment = to provide controlled condition of the containment
  - Terminate leakage of Ra media to the surrounding environment
- Secondary goals of SAMG
  - Minimize leakage of Ra media to the surrounding environment while fulfilling the primary goals
  - Provide for maximum possible operability of the equipment while fulfilling the primary goals

The symptom-oriented approach is applied consistently to severe accident management. The basic principle of this approach is such that the corresponding strategy for solution is selected based on the current development of the accident, which is defined on the basis of unambiguous symptoms (attributes). If, in the course of solution of the accident, the symptoms change and the employed strategy cannot be applied any longer, the structure of the strategy principles makes it possible to change the original strategy and to continue in the solution of the accident by means of activities determined by a different procedure or guideline that better suits the newly occurring conditions. Uninterrupted diagnostics of the conditions of the unit in the course of accident thus enables to respond correctly to possible changing conditions of the accident´s development and the interventions are thus always an optimal response to the given condition of the unit, which takes into account also off-site events and imminent risks.

In case of termination of activities according to the emergency procedures and transfer to SAMG, activities according to the Manuals for TPS are terminated as well and the follow-up management of activities is subject to SAMG.

Scheme of communication between TPS and the operating personnel for the use of SAMG



(Legend: Příkaz = Command, SAMG pro TPS = SAMG for TPS, Velitel HŠ = Commander of HŠ, Vedoucí TPS = Head of TPS, Technolog TPS = Technologist of TPS, Řízení zásahů = Interventions management, Dosimetrista = Head of radiation protection, Informatik = Head of information technologies, Skupina SAMG = SAMG Group)

Severe Accidents Management Guidelines for shutdown states (SAMG for shutdown states), 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 resolve beyond design basis events due to fuel damage in SFP, have not been completed for EDU yet.

For the maintenance of EOPs, SDEOPs and SAMG, updating thereof is performed on a regular basis consisting of knowledge based on training the use thereof 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 of the so-called “Maintenance program”.

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.

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 EDU was performed after the SAMG implementation. Training according the pre-prepared scenarios of the course of severe accidents with variants of different interventions was supervised by specialists from the Westinghouse company who are experienced in executing trainings in other locations all over the world.

With respect to the fact that the full scope operation training simulator is not designed to simulate a course of severe accidents, a simulation tool enabling to display the course of parameters, the behaviour thereof in time and space for all the analysed scenarios of severe accidents is being developed in cooperation with ÚJV Řež. This concerns one of the provisions arising from the Periodic Safety Review (see chap. 1.2.2). Visualization of the unit´s response to various interventions within the framework of the analysed scenarios of severe accidents will facilitate update and maintenance of SAMG and at the same time, it will be used for training and practical training of the personnel of the Dukovany nuclear power plant (in particular members of TPS) for the application of the SAMG procedures to manage severe accidents. The tool is based on animated display of the course of a severe accident in the reactor, the primary circuit and in the containment. The graphic representation will be interactive: depending on the selection, speed of the graphic representation will be optional as well as repetition of the selected sequences of the accident and the set of supplementary animated diagrams of the accident´s parameters.

6.1.1.6 Dependence on the functions of other reactors on the same site

In EDU location, there are 4 reactor units arranged in twin-units. The containments of the individual units within the twin-units are strictly separated from one another during operation and therefore no leak of atmosphere from one unit into the other one is possible. During refuelling modes 6 and 7 in one unit, the containment is open to the reactor hall, which is shared with the twin unit, however, it is hermetically separated from the containment of the twin unit. In the reactor hall, there are storage fuel pools for both the units. In case of accident during refuelling, it is necessary to resolve the issue concerning leak of radioactive substances to the shared reactor hall and the open containment of the affected unit.

The reactors are fully independent from technological point of view, however, a number of systems and auxiliary and supporting equipment are interchangeable. E.g. the electrical supply systems, circulation cooling water system, fire water system and similar systems are interconnectable between all the units.

For example, the essential service water system (ESW) has the aforementioned feature. On each unit, there are 3 independent systems to cool essential current consumers. The pumps to supply the individual units are located in separated boxes in a single building, are power supplied from the respective units and systems, however, they can be used by either of the twin units, which means that units 1 and 2 are serviced by common pumps, similarly as in case of units 3 and 4. it means that each unit can be supplied by 12 ESW pumps (see chapter 1.3.2.2).

The twin-unit arrangement using auxiliary systems enables replacement or replenishment of media in the ECCS tanks from the twin unit in case of emergency. In case, only one unit of the twin-unit is affected, it is also possible to use supplies of water in the passive bubbler trays of the twin unit, which means approximately 1,000 m3 of the H3BO3 solution.

With respect to the total number of 4 units in the location, arranged as neighbouring twin-units, and with respect to the independent character of power supply of the individual units from on-site as well as off-site power sources (including stand-by ones), it is possible to use power sources pertaining to one unit in case of SBO occurrence in the other unit.

The fuel storage pools are located in the reactor hall next to each reactor. The reactor hall of the twin-unit is interconnected. With respect to the common support systems (TM water cleanup system), it is possible to connect the units in case of emergency make-up. Location of the fuel storage pools outside the containment enables a simplified access in case of emergency make-up by other emergency means (fire fighting equipment, etc.).

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 emergency operation 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).

In the EDU location, there is the On-site Fire Rescue Brigade (FB), which is equipped with relevant fire-fighting equipment and which is trained to intervene at any place of the location. The FB 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. The fire-fighting equipment and the intervention staff (48 fire-fighters in 4 shifts) are located in the civil structure of the Fire Station where no direct effects of extreme natural hazards can be expected, however, seismic resistance of the building concerned has not been assessed yet. In case of damage thereto, the Fire Brigade intervention might be limited.

For the purpose of upgrade of HZSp of the Dukovany NPP, an emergency plan, which forms a part of the off-site emergency plan of the Dukovany NPP, has been prepared and on the basis thereof, the professional brigades of the Fire Rescue Service of the Czech Republic which are an integral part of IZS, would be able to provide for further effective material and personal assistance with the time necessary to reach the location in the range of 10 - 60 minutes depending on the fire rescue service team dislocation.

The FB mobile pumps (pump delivery pressure is 0.8-1.2 MPa, flow rate 120-150 t/h) can be used for the demineralised water make-up directly into the SG in an alternative manner. Within the framework of the design completion, there are points of supply ready to interconnect this equipment with the technology. The alternative way of SG make-up is described in EOPs and has been tested in practice several times and the capacity of this equipment to secure basic safety functions has been verified. The realistic time necessary for actual water supply to PG by means of a mobile pump measured from the request for activation of HZSp is approx. 20 minutes. In case of loss of the SG level instrumentation system and other data, tables for the possibility of optimal demineralised water make-up have been prepared, which specify what flow rate of demineralised water is necessary at the respective counter-pressure in SG so that the flow rate of the supplied demineralised water matches the steam extraction though steam dump to the atmosphere. Upon black-out on all four units of EDU, however, the capacity of the necessary fire-fighting equipment may be a certain limitation (as the emergency plans for the SG supply in two units by means of one pump at the same time have not been prepared yet).

Among other alternatives, there is the use of fire-fighting equipment to make up for the lost coolant and to maintain the fuel temperature in SFP. EOPs contain this possibility of alternative SFP make-up, however, specific procedures for interventions on site have not been prepared yet.

In addition, SAMG takes into account the use of portable power packages to control some valves directly from the distributors; the specific procedures for interventions on the spot have not been prepared yet.

In addition, pursuant to the respective legislation, it is also possible to deploy other basic as well as other services of IZS (ZZS, Police of the ČR, Armed Forces of the ČR, ...). Depending on the degree of extraordinary event in the nuclear facility, the individual services operating in the rescue system fulfil tasks aiming at mitigation of the extraordinary event on the affected equipment or minimization of consequences thereof. The tasks may be executed in the territory of the Dukovany NPP, in the emergency planning zone or outside the emergency planning zone.

6.1.2.2 Provisions for and management of supplies (fuel for diesel generators, water, etc.)

The reserve of fuel oil in the operating tank for each DG is sufficient for a period of at least 6 hours (4.5 m^3 of fuel, max. load consumption 0.7 m^3/h). For each DG, one, interconnected pair of storage tanks has been determined, which contains a minimum reserve of 110 m^3 of fuel. Repumping of fuel oil from the storage tanks into the operating tank proceeds automatically in case of drop of level on the operating tanks. The pumps to transport fuel are power supplied from the respective DG. The total reserve of fuel oil, which is 114.5 m^3, is sufficient for the operation of one DG for a period of at least 144 hours (in fact approx. 160 hours), which means 6 to 7 days without the necessity of external fuel supply.

Note: Other fuel for DGs could be available by repumping from other DGs (which are, for example, out of order) by means of the so-called reexpedition pumps, namely provided that these pumps are power supplied (from a system outside the scope of the design). While considering a long-term operation of always just one DG on each unit, putting the reexpedition pumps into service would mean to have fuel for a period of 18 to 21 days at the disposal without the necessity to supply EDU with fuel oil.

For demineralised water make-up by means of optimal flow rate, the existing reserves of demineralised water in the 3 x 1,000 m^3 tanks for each twin-unit are available, which, according to the respective analyses, would be sufficient for 72 hours for all the 4 units. Together with the use of the coolant reserves in FWT, the coolant reserve for the SG make-up of all the four NPP units is sufficient for approximately 4 days. For the PG supply by mobile means, it is possible to use, in addition to the reserves of coolant in the tanks of demineralized water, the coolant from the cooling towers pools or other sources as well.

In case of loss of the raw water make-up system, in case not-secured power supply is available, it is possible to use the reserve of coolant in the decantation tanks in the amount of approx. 5 x 2,000 m^3 and the raw water supply in the gravity distribution reservoir with the volume of 4 x 2,000 m^3 to compensate the losses of ESW by evaporation.

As it follows from the analysis of the loss of ČSJ and ESW, EDU water systems provide for a reserve, according to the conservative approach (assuming only halves of CČSI and CČSII, the level in the cooling towers is minimum -2.55 m) of approximately 75,564 m^3 of water. This reserve is sufficient for 931 hours (approx. 39 days) of residual heat removal (operation of the ESW pumps) from the shutdown reactors without the necessity to supply water to the EDU systems.

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 NPP 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.

Monitoring of the radioactive media situation on-site as well as off-site is realized by means of radiological systems CISRK which, in the existing design, does not have seismic qualification on the level of maximal design basis earthquake and is located in the premises, which do not comply with the respective seismic resistance. Moreover, they are not power supplied from the secured power supply of category I. For radiation measurement, an alternative method by means of portable measuring devices will remain available.

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, EDU uses the RTARC software utilizing the current meteorological data, the meteorological data forecast, data concerning leak, the HVB dimensions, data concerning the land relief of the NPP environment and data concerning the radionuclides included in the radioactive media subject to leak. 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.

The RTARC system works with data from pre-calculated source term, which can be amended based on the currently measured values of leaks. The source terms assuming fuel meltdown in an open reactor and in SFP have not been finalized yet.

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 protecting measures including changing residence, control of food and water contaminated by radionuclides and control of animal fodder contaminated by radionuclides.

The safety 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 protecting 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 protecting measures in case of the extraordinary events´ occurrence relating to the time of 8 hours as of the extraordinary event´s occurrence.

	SAFETY 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.

The limits (see next page) 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.

On EDU, several shelters have been constructed. In one of the shelters, ECC 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 2,450 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 (protective clothing - TYVEK overalls, gaiters, gloves, respirator and one dose of iodine prophylaxis KI).

Iodine prophylaxis using the potassium-iodine tablets is performed in all persons, except for persons aged over 45 who have been identified as hypersensitive to iodine products or with a treated thyroid disorder. All the NPP employees and persons living in EPZ EDU are personally equipped with potassium-iodine tablets. The persons residing within EPZ EDU 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, instructions concerning initial protecting 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, the freeset system
- 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, periodic verification and testing is performed.

- Once every three months, for the functionality of technical equipment, systems and methods of activation of intervening persons for intervention management and performance.

- Once every six months for the functionality of technical equipment, systems and methods of warning of the personnel and other persons within the NPP premises.
- Once every 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.

For all the communication systems, the method of confirmation and verification based on periodic checks has been set. 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 selected mobile telephones, via OPIS HZS, have been set the priority of calls realized in the network with respect to extraordinary events management. EDU has an independent system, which is used to activate the hooters and to inform members of the Stand-by Emergency Response Organization. The system is autonomous and is not dependent on possible overload of the mobile telecommunications 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 TSC to all MCRs. 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. A design of the visual camera system for MCRs, TSC, shift supervisor is being finalized, including the closed circuit TV system´s completion.

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 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 (ECC, shelters, FB, SUJB, IZS, MCR 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 within EDU civil structures do not have stand-by power supply. The internal radio system is without stand-by power supply. The hooters in the civil structures have own accumulator batteries. The operation internal radio has a redundant power supply.

In case of a long-term SBO, a loss of power supply of EDU telephone central and telephone centrals of EDU off-site cooperating network workplaces might occur, with the exception of Power load dispatching centre Prague and Ostrava, 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 EDU (e.g. from EDA, or VE Vranov) is conditioned by cooperation (necessary connection) of several entities (ČEZ, ČEPS, E.ON).

In case of damage to the infrastructure, communication between the persons performing intervention and the control centre might be threatened as well as with the off-site centre of the public administration bodies (Crisis Centre SUJB, the Regional Crisis Centre, IZS, etc.) as the availability and capacity of the existing communication means is rather limited. 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 FB wireless stations to the Fire Brigade in Třebíč.

In case of SBO, the internal radio will be out of order (completion with UPS stations is planned). The operation internal radio should remain in order in case of SBO. The internal warning system: internal hooters (located in the civil structures) will be power supplied from own batteries. The hooters in the civil structures will be out of order (the solution is to replace the existing rotational hooters with electronic ones power supplied from accumulator batteries).

The transmission equipment between the individual centrals in the ČEZ ICTS network is controlled on a dynamic basis (selects a free transmission route automatically) and for this reason, the period of operation based on stand-by power supply sources depends on the location where loss of power supply occurs.

The active elements of the Duknet network are power supplied mostly from lighting switchboards (including the users PCs) and supported by UPSs. The central node at AB 1 is designed for a period of 2 hours.

The MPLS WAN ČEZ network, which provides communication between the data centre and the individual ČEZ locations, is redundant for a period of 1 hour in case of loss of power supply.

The control system of the internal warning system is deployed in two independent workplaces OED and EC, which are stand-byed by means of UPS.

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 to NPP, no crisis plans with the use of support or alternative technical equipment have been prepared. Possible use of support and alternative technical equipment would be resolved by means of OHO mechanisms. In case it is not possible to use the Emergency Control Centre for any reason, the stand-by centre shall be LRKO in Moravský Krumlov which disposes of limited amount of information to manage extraordinary situations, has been determined as a stand-by centre for the accident management.

In case of unavailability 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 ECC, 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, in case of imposition of protecting measures, from where they would perform the required interventions on the technological equipment or provided operational support to FB with respect to recovery and rescue works depending on instructions provided by shift supervisor or EC.

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, telecommunication means for communication with the EC workplace. 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 location, 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

The MCR and ECR rooms are located in rooms adjacent to the containment. This section might be exposed to radiation at higher pressures and at high doses inside the containment occurring at the same time or in case of extensive leaks of fission products from the containment. Equipping the MCR and ECR with the filtration ventilation systems has not been completed yet. In substantiated cases, shift supervisor, safety engineer or unit supervisor may decide on the transfer of the MCR staff to ECR. Usage of breathing apparatuses in MCR falls within the competence of unit supervisor.

Prior to completion of making MCR and ECR more resistant, evacuation of the staff upon the EC commander´s order based on the assessment of radiation situation is assumed while adhering to the criteria imposed by the intervention instruction (subsequently, only short-term presence to perform interventions would be assumed).

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

All necessary activities in case of the extraordinary event occurrence would be managed and performed from the protected areas. Activities according to SAMG are managed by members of TSC and EC gathered in the shelter intended for the Emergency Control Centre. Remote activities to implement the respective strategies would be performed by the operational management staff from MCR or ECR. Local interventions and possible repairs of the equipment would be performed in the respective rooms of the reactor building, turbine building and off-site civil structures by the staff gathered in the Operational Support Centre, which is located in a shelter within EDU premises.

6.1.3.6 Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods)

The NPP as well as OHO staff is qualified and trained to use EOPs and SAMG in case of earthquake and floods. However, the staff is not sufficiently trained to use the support and alternative technical equipment, for the use of which the respective procedures, manuals and plans have not been prepared yet. Within the framework of the shift (OHO) or POHO, deficiencies concerning the number of staff members necessary to extenuate consequences of a beyond design basis event have not been identified yet.

In case of extensive damage to the infrastructure and long-term unavailability of the location (destruction of buildings, damage to communications, etc.), the altering staff might not be able to get to the location. In this case, the required activities would have to be performed by the staff present at the time of occurrence of the event. Altering would be resolved on an operational basis in cooperation with the state administration bodies (IZS, the army, etc.).

It is likely that it would not be possible to use the emergency preparedness shelters or EC workplaces, or the workplaces of TSC, which are located under civil structures with insufficient seismic resistance and which are not protected against flooding. Unavailability of TSC could be partially resolved by its functioning from the emergency control room (detailed instructions are not available yet). Possible unavailability of the shelters would be resolved on an operational basis by evacuation of the staff out of the location.

Another difficulty might be availability of information concerning radiation situation within and on the boundary of the EDU premises. All the currently operated systems radiological systems (CISRK) are not in the seismic-resistant version, or parts thereof are located in building with insufficient seismic resistance. The radiation monitoring could be performed in an alternative manner using hand portable measuring devices.

In case of floods and long-term unavailability of the location, the altering staff might not be able to reach the location operationally. In this case, the required activities would have to be performed by the staff present at the time of occurrence of the event. Altering would be resolved on an operational basis in cooperation with the state administration bodies (IZS, the army, etc.).

It is also necessary to take into account local restriction of passability of the communications in the neighbourhood of NPP due to mudslide from the surrounding fields. The network of access communications and bridges across the watercourses in the valleys surrounding the plant, however, is dense enough that it is likely that access to the plant will be possible.

In case of unavailability 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 ECC, the possibility to use the information centre building).

6.1.3.7 Unavailability of power supply

The assessment of power supply unavailability has been provided in chapter 5.1.

The limited capacity of the ZN accumulator batteries of category I might not be sufficient to enable all interventions in the early stage of a severe accident and might prevent some measurements. Even long-term power supply of essential current consumers, including measurements necessary for SAMG might be difficult. These measurements are concentrated in PAMS, which is power supplied from ZN category I. The staff performing intervention and participating in the management and performance of handling works might not have all the necessary information at their disposal

In the frame of SAMG there is incorporated using of portable devices for local control of some selected valves directly from the electrical distributors. Fire brigade personnel has three small dieselgenerators at disposal (voltage 3x380V):

GEKO BSKA (5,5 kW), MITSUBISHI 4200 (3,6 kW). FORMULA 6000 T (3,6 kW)

Until full run out of the batteries, only emergency lighting would be functioning. Loss of lighting could contribute to aggravated orientation of the staff and thus to extension of the time necessary for handling works.

6.1.3.8 Potential failure of instrumentation

Most of the required information concerning the state of components and values of parameters necessary for the severe accidents management is available in PAMS.

All the systems are qualified for design basis accident and post-accident conditions. They are not qualified for severe accident conditions, however, in a number of cases their measuring scope is designed to comply with requirements concerning management of initial stages of severe accidents.

- Temperatures at the outlet from reactor core within 1200 ℃
- Temperatures in the loops within 400 ℃
- Overpressure in the box within 450 kPa.
- Measurement of hydrogen concentration within 10 %.

A limited set of parameters is used for the diagnostics of the accident situation and for the verification of the selected strategies implementation. For the verification of the parameters, measured values of the selected quantities from the standard instrumentation are used. For each parameter, several quantities have been determined, by means of which it is possible to verify the respective parameter (as to its value, trend). A direct measurement of the required parameter and one or several measurement of the alternative quantities, based on which it is possible to derive the value of the trend of the required parameter, are always used. In some cases of a severe accident, it is impossible to acquire value or trend of the required parameter based on directly measured values either due to unavailability thereof or due to non-existence of the respective parameter measurement. In such cases, computational methods (simple diagrams of the parameters´ dependence) are used for determination of the required parameter. Inputs of the computational methods are either directly measured values or pre-determined defined values.

The ability of the instrumentation´s survival under conditions of severe accident is not known, however, it is assumed that it is robust enough to resist the sever accident conditions for at least some time.

Measurement of hydrogen concentration depends on the oxygen volume in the measured area and will be rather inaccurate in case of high concentration of hydrogen or high consumption of oxygen. Nevertheless, it provides for basic information concerning the hydrogen-related risk, in case the measured concentrations will be much lower than the real ones due to low concentration of oxygen, which means a low risk of ignition at the same time.

PAMS does not provide for information concerning the radiation situation monitoring, or the SFP condition for the time being. However, the measurements are performed by standard means. On the other hand, provision of power supply for PAMS at SBO still remains the main problem concerning the measurement availability.

Within the framework of the design of EDU equipment qualification with respect to the selected accident events, the required thermohydraulic and radiation parameters of the environment have been determined. For each space subject to monitoring and for each thermohydraulic parameter subject to monitoring, the qualification envelope curve of the respective parameter has been created based on all known modes of behaviour.

In case of severe accident, conditions in the containment will be unfavourable on a long-term basis and will reach the level of hard environment conditions. Above all, it is necessary to take into account extremely high dose rate, which may reach up to extreme values due to locally deposited aerosols and lead to damage to the sensor. Hydrogen combustion generates special conditions. Temperature of the atmosphere may exceed 1,000 ℃ for several seconds and pressure may exceed the design basis overpressure of 150 kPa depending on the fire intensity. Nevertheless it is likely that a larger part of the instrumentation would survive the fire as the thermal capacity of the walls and the equipment is much higher than the atmosphere capacity. radioactive media measurements realized in the radiological systems are able to continue working in hard environment on a long-term basis, even in high radiation assumed during severe accident. The other measuring devices are less robust and resistant than for example the spray system as damage to cable insulation may occur due to hydrogen combustion. Complete information concerning behaviour of the impulse tubes and measurement of level exposed to high temperatures during hydrogen combustion on a short-term basis are not available.

All the aforementioned effects may lead to deterioration of the measurements accuracy. Above all, it is necessary to take into account the “unreliability” of the details concerning temperature. It is better to rely on the details concerning pressure, which will be relatively balances on a long-term basis and the differences of which provides information, for example, on the functioning of the vacuum bubbler system. Dose rate is measured by qualified instrumentation and the data may be used directly (while taking into account deposition of aerosols on the detectors). In addition, SAMG rely on measurements of doses outside the containment, by the interpretation of which, it is possible to estimate the leak route.

6.1.3.9 Potential effects from the other neighbouring installations at site, including considerations of restricted availability of trained staff to deal with multi-unit, extended accidents

No risks due to effects of industrial installations on EDU location have been identified. No such installations are located in the area surrounding the plant. The shift staff´s capacity for the initial activities is sufficient, however, for long-term management of accident conditions, the staff assignment shall be subject to special regime (altering in exposed workplaces, organization of rest, meals and conservation of available resources).

6.1.4 Conclusion on the adequacy of organisational issues for accident management

The purpose of the accident management on NPP is to provide for the level 4 of the in-depth protection (to extenuate consequences of the accident occurrence). This level is followed by the NPP emergency preparedness, such as the level 5 of the in-depth protection (to extenuate consequences of accidents accompanied by radioactive media leaks). The functional and interconnected system of accident management and emergency preparedness is provided on EDU by means of a robust set of measurements of personnel, administrative and technical character.

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 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 EDU with the assistance of the technical support centre.

In case of an abnormal event, all the necessary activities would be managed and performed from protected places. 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 even in case of activity leak into the environment. Remote activities to implement strategies would be performed by the shift personnel from MCR or ECR where the occupancy project for the control centers is being finalized. Local activities and possible equipment repairs in the relevant areas of the reactor hall, turbine hall or external buildings would be provided by intervention teams at the operational support centre.

The concept of the accident management system at EDU 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 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 unfailed 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 the event 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 organisation of abnormal event management (including severe accidents) is defined in the on-site emergency plan approved by SUJB (State Office for Nuclear Safety).

In 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 at the EDU 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 EDU 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 specific scenarios of severe accidents.

6.1.5 Measures which can be envisaged to enhance accident management capabilities

Despite the robustness of the system of accident management and emergency preparedness there are still areas for improvement.

In the area of administrative management, this concerns in particular the Manuals of severe accidents management for shutdown states (SAMG for shutdown states), which have not been finalized for EDU yet. Nevertheless, for the purpose of EOPs, SDEOPs and SAMG, the so-called "Maintenance program", including the update thereof, is performed on a regular basis.

In the personnel area, problems concerning availability of the location or the ECC 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.

The plans to make the existing system even more resistant aim at assessing the readiness to manage extraordinary events from the Stand-by Emergency Centres (in case of inaccessibility of the affected location) and at periodic reviewing of the nomination of the best POHO staff as far as their specialist qualifications are concerned.

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 control emergency preparedness centre , 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 regulations / procedures / manuals to manage the selected beyond design basis states and accidents of NPP (SAMG for shutdown states, SAMG for damaged fuel in SFP, EDMG, …) with a view to securing cooling and heat removal from the reactor core and SFP and eliminating leaks.

3. Improve the standard of personnel training in the area of severe accident management (use of the simulation tool to display the behaviour of parameters, phenomena and behaviour of the unit with respect to specific scenarios of severe accidents).
4. Additional technical measures to provide for non-technical support functionalities (accessibility of the civil structures, availability of the fire-fighting equipment, provision of ECC and shelters, physical protection systems …).
5. Alternative means to secure long-term functional communication between all components of the accident management system.

The opportunities for improvement of the in-depth defence during events are provided in the following table.

Some of the provisions (designated as “PSR finding” in the note) would be implemented even without this goal-directed review, which has confirmed in its outcomes the effectiveness and correctness of the previously taken decisions to implement provisions aiming at upgrade of the original design. A more detailed description of the corrective actions is provided in the annex.

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
Habitability of BD during severe accident	Provision of habitability of BD	II	PSR finding
Habitability of the shelters during severe accident	Oxygen regeneration in the shelters	II	PSR finding
PAMS	Completion with measurement concerning Ra situation and the BSVP´s condition	II	
Regulations	Prepare "shutdown SAMG" for outage / severe accident in BSVP	I	PSR finding
Emergency preparedness	Provide alternative means for warning and informing the EDU´s staff and the members of the public in the ZHP zone	I	
Regulations	EDMG guidelines for the use of alternative means	II	
Emergency	Provide for the functioning of	I	

preparedness	the emergency response teams in case of HŘS´s unavailability.		
Staff	Improve the standard of training and practical training of TPS in the area of severe accidents	I	PSR finding - A simulation tool for display of severe accidents is being implemented
Emergency preparedness	Prepare agreement with off-site services (IZS, Armed Forces of the ČR) and near NPPs. Organizational provisions.	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/a number of pressure tubes (including last resorts to prevent fuel damage)

The basic cause of severe accidents is insufficient removal of heat released from fuel in reactor core. Damage to reactor core refers to local exceeding of the clad temperature of 1,200 ℃ when the steam-zirconium reaction develops. With respect to impossibility to measure this parameter, a setpoint for the SAMG transfer to the value of temperature at the outlet of reactor core in the amount of 550 ℃ has been determined. Exceeding 1,200 ℃ in a more extensive area leads to intensive steam-zirconium reaction, which is exothermic. In this manner, much larger amount of heat is released fast compared to decay heat and this heat will contribute to development of the accident as it is mostly accumulated inside reactor core.

Recovery of heat removal from reactor core on the part of II.O using alternative means is performed in EOPs, which means even prior to transfer to SAMG. While ECCSs are being recovered, activities relating to depressurization of I.O are being performed with a view to enabling the low-pressure pumps injection into I.O.

Two permanent ways of stopping the reactor core cooling loss development into a severe accident:

- Recovery of heat removal via SG (alternative SG make-up by means of low-pressure sources, including feeding SGs using the means pertaining to FB).
- Removal of heat by I.O make-up and its discharge through the escape opening in the primary system during LOCA or by PRZR SVs or PORV (Feed & Bleed).

The EOPs include alternative strategies as well:

- Depressurization of the primary system, or cooling on the part of II.O, which may lead to the assertion of hydroaccumulators or even low-pressure emergency or alternative resources.
- Recovery of operability of the high-pressure safety injection systems or alternative high pressure systems.
- Utilization of the remaining coolant in the loops by forced start of RCPs even at the price of destruction thereof.

6.2.2 After occurrence of fuel damage in the reactor pressure vessel

Under a conservative approach, we can associate fuel damage with the beginning of steam-zirconium reaction relating to massive generation of hydrogen, which precedes the beginning of the reactor core geometry loss. The symptom of the reactor core damage by meltdown, in addition to ever increasing temperature, is in particular the increase of the hydrogen concentration in the containment. With respect to the speed of the hydrogen generation prior to the loss of geometry, the existing recombination devices would not have to be fast enough to manage the hydrogen concentration. However, there is still a time allowance (in the order of tens of minutes) for possible safe ignition of hydrogen in the initial stage.

A typical time as of entering SAMG until RPV integrity damage by the effect of the reactor core RPV meltdown is approx. 7 hours provided that all ways of coolant supply to the vessel failed.

The principal strategy, which is applied in SAMG in this stage of accident, is reduction of pressure in the primary system due to reduction of hydrogen generation, however, in particular due to elimination of creep damage of the vessel bottom and high-pressure blowdown of the melted core from the vessel. In compliance with the generic manuals, the required value of pressure in I.O. is less than 2 MPa, specifically for EDU units, the value of 1 MPa has been determined. The effect known as the “direct containment heating” in the whole volume is not relevant or likely in case of VVER-440. However, load applied on the wall of the reactor cavity due to high overpressure of gases from the primary system represent risk in case the opening after the vessel failure is large enough and contributed by “direct heating” of the cavity atmosphere from particles of the melted fuel.

Reduction of pressure in the primary system is one of the highest priorities aiming in particular at preventing blowdown of debris from the vessel under high pressure as described in chapter 6.3.1. Fuel damage due to steam-zirconium reaction under high pressure also contributes to generation of hydrogen and therefore it is necessary to reduce the pressure long before the risk of the vessel´s bottom´s failure occurs. The PRZR SVs and PORV can be used for this purpose.

Between the beginning of the reactor core damage and the vessel failure, several partial stages corresponding with gradual loss of geometry, creation of the melted core pool on the load-bearing plate and its failure and the advance of the debris onto the vessel´s bottom must be distinguished. With respect to the large amount of water in the lower part of the vessel, however, failure of the bottom appears in several hours, the presence of water as such cannot prevent the failure as the accumulating debris will create configuration, which cannot be cooled down.

The strategy of the heat removal recovery is resolved in SAMG by means of [SAG-2] (depressurization) and in particular by means of [SAG-3] (I.O make-up). In this stage of accident, it is not possible to use the I.O cooling on the part of II.O, as it is necessary to supply coolant directly into the reactor vessel.

The sooner water is supplied, the higher the chance to stop the accident and to maintain the melted core in the vessel is. In addition to the case when most debris is already on the bottom of the vessel under the unfavourable configuration, there is still a certain chance to prevent the TNR´s bottom´s failure. Therefore, the SAMG guidelines recommend starting to supply water at the moment when the power supply is recovered in the quantity larger than the minimum flow rate necessary for the reactor core reflooding. The flow rate was determined as the flow rate which is evaporated by the residual heat of reactor core according to computational method used in SAMG

The risk of the vessel´s failure would be reduced in a considerable manner by implementation of the strategy based on cooling the vessel from outside by means of reflooding the reactor cavity. Success of this strategy has been proven in an analytical manner. Within the framework of the technical solutions implementation concerning modification of the ventilation supply pipelines to the reactor cavity, the feed openings from the floor of the SG box enabling reflooding of reactor cavity room after completion of the entire design have been realized already.

6.2.3 After failure of the reactor pressure vessel

In case of failure to stop the accident within the reactor vessel, the failure of its lower part would be unavoidable, including interaction of the melted fuel with concrete. The main consequences of this stage of the accident could be as follows:

• Additional hydrogen generation from non-oxidized Zr, steel in the debris and the concrete armouring.

• Breakthrough of the melted core out of the reactor cavity.

Hydrogen generation during interaction of the melted core with the concrete of the reactor cavity bottom is not so fast as in case of the cladding oxidation. It is slower by two orders than the hydrogen generation in the reaction between steam and the zirconium cladding.

Breakthrough of the melted core out of the cavity wall is more severe than breakthrough out of the cavity bottom from the following reasons:

• Breakthrough of the melted core is faster in radial direction than in axial direction.

• The wall 2.5 m is thinner than the bottom 3.1 m.

• The cavity wall represents the containment boundary; through the bottom, debris will go through the base plate (bottom layer) where fission products could be filtered better.

Damage to the vessel´s bottom would result from thermal creep. At a higher pressure inside the vessel, this may happen at a lower temperature of the bottom and even before the melting of debris on the bottom. Localization of damage depends on pressure; at a higher pressure it is in the place of highest tension on the vessel´s bottom, at a lower pressure, in the place of attenuation of the vessel in the cylindrical part.

After damage of the vessel due to TNR, relocation of materials from the vessel would occur and gradually a layer of debris with such a dense arrangement would develop that it would be impossible to cool it down as melting of the debris would occur together with interaction with concrete even under the possible layer of water, which is insulated by the film boiling. With the melting of the debris, their foaming with steam and CO released from the concrete during interaction with the debris and hydrogen and CO released from the melted core after its oxidation.

On the surface of the melted core in the cavity, a solid crust would develop, which would prevent heat removal from the surface of the melted core even in case it is under water.
With the VVER-440 reactor, the door in the cavity was protected by the solid debris or by the crust against contact with the liquid debris for a certain time. Nevertheless, the crust has low thermal conductivity. With respect to the melted core foaming and low melting temperature of steel, the most likely scenario would be the meltdown of the lower part of the door and the pouring of the part of the melted core down to the second door along the corridor, which would fail after a certain time as well. In any case, it is not possible to exclude minor damage to the containment shortly after the failure of the vessel´s bottom due to the failure of the rubber sealing of the door.

By reflooding the debris in the cavity, the door might be protected. Even in case of damage to the sealing, there is still the sealing of the outer door, which may prevent leak of water and thus protect the door. This method of the door protection has not been analysed yet; the idea is based on the informed guess.

If the measures to protect the doors against failure are implemented, breakthrough of the melted core out of the cavity wall could occur no sooner than in 4 days as of the moment of the vessel´s bottom´s failure. This means extensive late damage to the containment. Concentration of fission products in the containment atmosphere at that time would be low.

The strategy of the melted core cooling in included in the [SAG-5] cavity reflooding. The current configuration of the plant provides the possibility to reflood the cavity by overfall, which would require water from two ECCS tanks and the bubbler trays. Therefore the manual assumes discharge of the bubbler trays, including a check of closure of the waste-water disposal system of the box to prevent further losses. The spray pumps or the low pressure auxiliary systems pumps could be used for pumping water from the ECCS tanks. The strategy considers using the water reserve from the twin unit as well.

The main benefit of the strategy of reflooding debris in the reactor cavity is then cooling the steel door and capturing fission products released during interaction between the melted core and the concrete.

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 PRZR SVs are the basic design basis instrument for the provision of depressurization of I.O. Another option is to use PRZR PORV (the flow rate cross section is limited) and SG depressurization.

6.3.1.2 Operational provisions

As early as in EOPs, there are strategies to maintain the safe condition on a long-term basis, which are based on the controlled cooling down and in particular depressurization of I.O. One of the main priorities of SAMG is to prevent damage to TNR by the melted reactor core melt-through at high pressure; the activities are commenced as early as in EOPs and subsequently immediately after transition to SAMG due to the risk of increase of the hydrogen generation.

If early depressurization of I.O. fails to be performed, which means prior to massive steam-zirconium reaction, there is still sufficient time to prevent the vessel´s failure under high pressure. The effects associated with hydrogen generation, such as combustion or detonation, however, may affect the situation inside the containment and thus the PRZR SVs and PORV control.

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

In the initial stage, the containment integrity is most threatened by extensive fire or detonation due to hydrogen, followed by failure of the double door in the reactor cavity. In the late stage of the accident, the breakthrough of the debris out of the cavity contributes as well. The threat to the containment due to hydrogen may occur after the beginning of the reactor core damage during the steam-zirconium reaction. Due to the large surface of the cladding and exothermic character of the reaction, the hydrogen generation is very fast, in the order of 0.5 to 1 kg/s. With respect to the speed of the hydrogen generation prior to the loss of geometry, such amount of hydrogen is not manageable using the existing recombination devices. The hydrogen generation would continue in the late stage of the accident as well during the reaction of the melted core at the bottom of the reactor cavity at the speed slower by two orders (less than 0.01 kg/s). From the point of view of the containment´s integrity´s risk due to hydrogen in the late stage, the risk would certainly increase, however, provided that the containment is still in one piece at that time. It is very likely that combustion of extensive quantities of hydrogen could be burned down as early as in the early stage; in a worse case, fast combustion or detonation might occur, which would lead to irreversible damage to the containment, and hydrogen could leak freely. Possible blowdown of the reactor core debris from the reactor vessel is also a considerable source of the hydrogen ignition.

Intensive hydrogen generation occur in approx. 30 minutes after the temperature of the gas at the reactor core outlet has exceeded 550 ℃. The behaviour of the steam-zirconium reaction is significantly more intensive at high pressure and therefore, one of the SAMG´s requirements is the instruction to depressurize I.O. The most extensive portion of hydrogen is generated during one hour prior to the melted fuel relocation.

The hydrogen released to the containment could lead to dangerous explosive concentrations in particular in the bubbler cavity where steam concentration is lower due to its condensation on the surface of the trays. The containment integrity could be put at risk so sooner than upon burning down hydrogen in a large volume of the entire containment.

The major sources of hydrogen in case of severe accident result from the steam-zirconium reaction on the surface of the cladding and shells in the reactor, or from the reaction of steam released from the concrete in the reactor cavity with metals contained in the debris

The EDU units´ containments are equipped with the systems of disposal of post-accident hydrogen designed for the design basis accidents only. For the LOCA design basis accidents when only very small amount of hydrogen is generated, disposal thereof is performed by means of 17 recombination devices located in the containment. Really effective disposal of the accident-resulting hydrogen would require an implementation of a highly-efficient system of hydrogen disposal. Upgrade of the EDU´s design in the area of severe accident management was decided on after execution of the Periodic Safety Review in 2006 (see chap. 1.2.2). In the final stage of the preparation, the design of construction of the system of efficient recombination of emergency hydrogen able to manage hydrogen produced according to the worst scenario (from the point of view of hydrogen production) of severe accident. The analyses have proven that such a system consisting of efficient recombination devices (PAR) combined with ignition devices provided that the spray system functions could reduce the risk of flame spreading to some partial small volumes and eliminate the risk of development up to detonation.

6.3.2.2 Operational provisions

The risk endangering the containment integrity due to hydrogen is resolved by SAMG, either on the principle of purposeful ignition or the containment´s inertization. Every random or intentionally induced combustion of hydrogen or its recombination decreases the oxygen concentration in the atmosphere and thus reduces the future risk resulting from hydrogen. Combustion or recombination of approx. 700 kg of hydrogen is sufficient for complete consumption of oxygen in the part of the containment without air traps. Additional hydrogen generated in particular during interaction with the concrete thus merely increases pressure in the containment, however, does not contribute to the hydrogen-related risk (as no oxygen is available).

Ignition assumes creation of spark using the electrical equipment inside the containment. In the manuals, there is a list containing equipment that the MCR would try to handle with (to change the position of the valves) with a view to generating sparks. The use of equipment located in lower altitudes would facilitate combustion. The use of the electrical equipment might not be fully efficient. Most of such equipment is in insulated anti-spark version. However, there is only a limited time interval as in case of fast hydrogen generation its concentration will be too high in a short time.

Discharge of nitrogen from hydroaccumulators could be used for the containment´s inertization, under the present state of the design the more efficient means of inertization is usage of steam, the hydrogen risk is merely postponed until even higher hydrogen concentrations. Above all, ignition of the hydrogen is likely from the recombination devices if the hydrogen concentration exceeds 10 % in the place of their installation. The existing recombination devices thus do not manage the hydrogen-related risk in case of a severe accident as they are able to remove only several kg of hydrogen in the early phase of the accident.

In case of the I.O depressurization prior to damage to reactor core (which is already performed within EOPs) and continuation of this procedure after reactor core has been damaged, the risk of detonation is postponed and confined in the bubbler cavity only.

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 design basis functionality of the containment is to prevent leaks of radioactive media to the environment, or to extenuate radiation consequences of an accident for the surrounding environment. The containment is the last barrier against the leak of radioactivity and is independent of the other barriers. The containment´s functionality is secured by its construction and structure, which is certainly able to resist the design basis overpressure of 150 kPa and is very likely to resist approx. double the overpressure.

The design basis functionality of the containment is secured in two ways:

1. By the use of the isolation QOV on all pipelines passing through the containment´s wall, by the use of hermetic openings and penetrations of all pipes and cables going through the wall.
2. By minimization of the leaks due to restriction of the internal overpressure duration with subsequent generation of underpressure compared to the ambient environment.

The system of suppression of pressure in the containment consists of two parts:

- The vacuum bubbler system containing passively functioning bubbler trays, which condensate steam and subsequently provide for passive spraying of the containment. Uncondensed gases and air in the containment environment are captured in the air traps, which are subsequently automatically separated from the containment environment.
- The spray system with three active spray pumps.

Cooperation of the two systems guarantees the creation of underpressure in the containment and complete elimination of leaks to the surrounding environment.

The right functioning of the bubbler condenser, which is important for the fulfilment of the VVER 440/213 containment´s safety function, was reviewed within the framework of the PHARE/TACIS PH2.13/95 project “Experimental Qualification of the Bubbler Condenser”. Tests, experiments on the unique equipment modelling the PG boxes and VBK at a scale of 1:100 and finally analyses have proven that the vacuum-bubbler system for the VVER 440/213 nuclear power stations (Paks, Dukovany, Bohunice and Rovno) is able to withstand the induced loads and to maintain its functionality. This concerns principle equipment limiting the maximum pressure in the course of accidents with extensive leakage. It provides for maximum contribution to reduction of pressure up to underpressure early after occurrence of the LOCA accident with extensive leakage and thus it prevents release of radioactive media to the surrounding environment. In case of development in a severe accident, it is not possible to maintain underpressure in the containment on a permanent basis, however, as it follows from the respective analyses, it is possible to guarantee the minimum overpressure and leak of radioactivity less than 0.1 % of volatile fission products with the exception of noble gases. In case of hypothetical failure of the active sprays, the bubbler system should provide for lower pressure in the containment than in case of the full-pressure containment and leak to the surrounding environment is less than 1 % of volatile fission products with the exception of noble gases. The vacuum bubbler system thus partially eliminates the tightness of the containment which is lower by up to two orders than the one of the full-pressure containment.

This is applicable to the containment the integrity of which was maintained; after its loss it is necessary to take into account very high leaks of radioactive media to the environment, which might reduce the functioning active spray system partially.

The threat to the DUKOVANY NPP VVER-440/213 containment due to overpressure of gases is very low (with the exception of a short increase during the hydrogen combustion). It follows from the undermentioned facts:

• The vacuum bubbler system condensates steam and creates underpressure in the containment at the beginning of the accident at the price of certain pressurization of its part - the air traps.

• The total volume of the containment, including the air traps compared the residual output is very extensive, approximately 50,000 m3.

• The rather high operating leakage of the containment amounting to several per cents of the gas weight/day at the design basis pressure contributed to pressure reduction. As the leakage is likely to result from small cracks in the concrete, aerosols, including water fog, may cause their blocking.

The pressure of 250 kPa (overpressure of 150 kPa) is the design basis pressure when damage to the containment is not likely. According to strength calculations for NPP, at the overpressure of approx. 290 kPa, the probability of the loss of the containment´s integrity is 5 %; overpressure of 350 kPa means probability in the amount of 50 %.

The results of the analyses concerning the likelihood of the loss on the containment´s integrity due to hydrogen overpressure show that after approx. 4.5 days, when the debris break through the cavity´s wall, the overpressure in the containment would be approx. 120 kPa, based on the guess, and if the cavity´s wall did not fail, the overpressure would reach the design basis overpressure in approximately 5 days. The leakage blocking affected the pressure behaviour in a considerable manner; without it, the maximum overpressure in 4.5 days would approx. 60 kPa.

Pressurization of the containment using steam in case of the heat removal and the spray system failure may be theoretically faster that pressurization with hydrogen. Such a situation, however, does not occur in practice. In case of heat removal failure, the loss of water and discontinuation of steam generation is likely.

6.3.3.2 Operational and organisational provisions

The strategy of pressurization prevention is described in the SAMG manual, Pressure control in the box, which is applied as early as at the overpressure of 10 kPa. The purpose of the aforementioned is to eliminate higher leak due to the existing leakage rather than to eliminate future threat to the containment due to overpressure. This corresponds with the application of only the containment systems based on heat removal such as the spray system or the recirculation ventilation while maintaining the pressure limit of the containment.

The SAMG manual “Pressure reduction in the box”, in addition to the aforementioned systems, assumes the use of ventilation systems for the controlled venting based on systems equipped with aerosols and iodine filters (which are not, however, designed for this purpose). Leak of radioactive media in this case would be limited as a part of noble gases would remain entrapped in the air traps.

The strategy aiming at preventing the containment overpressurization are applicable with the existing equipment and on an analytical basis it has been proven that it is not necessary to develop some other equipment such as a special system of filtered ventilation in case of severe accidents.

6.3.4 Prevention of re-criticality

6.3.4.1 Design provisions

For VVER-440, the risk of boron dilution in the advanced stage of an accident is lower than for the PWR reactors. With respect to the use of the control rods insertion together with ejection of a part of the fuel assemblies (37 of 349) from reactor core, the reactivity would be lower even in case of meltdown and relocation of the control rods.

6.3.4.2 Operational provisions

In the state of the reactor core sub-criticality risk in the prevention stage (EOPs), increased concentration id required in particular due to failing insertion of control rods and not due to compensation of the positive reactivity input resulting from the temperature decrease during cooling down.

After the loss of geometry, the issue concerning boron dilution does not exist. The geometry created by debris inside the reactor or in the cavity under the reactor is deeply sub-critical under all circumstance even in case of reflooding with non-borated water.

6.3.5 Prevention of basemat melt through

6.3.5.1 Potential design arrangements for retention of the corium in the pressure vessel

The VVER-440/213 design is advantageous from the point of view of retaining the melted core inside the reactor by means of its cooling even if the original design did not consider this measure. Above all, the residual output of the reactor is too low, which guarantees low heat flows on the external surface of the vessel in the area of nucleate boiling with an extensive reserve within the boiling crisis. The vessel does not have any penetrations in the lower part. The reactor cavity counts among the lowest places in the containment and in case of loss of the emergency cooling water, discharge of the bubbler trays is sufficient for reflooding thereof.

Upgrade of the EDU´s design in the area of severe accident management was decided on after execution of the Periodic Safety Review in 2006 (see chap. 1.2.2). On the EDU units, some modifications aiming at cooling the vessel from outside have been implemented. In particular, the waste-water disposal system is closed at the bottom of the cavity and the supply of the ventilation to the reactor cavity room has been modified in a manner allowing equipping it with the filling valves. What remains are certain modifications of the insulation in the lower part of the vessel not to prevent access of water to the vessel and minor modifications in the lower part of the cavity room (screen) and in the upper part (overflow onto the floor of the box).

6.3.5.2 Potential arrangements to cool the corium inside the containment after reactor pressure vessel rupture

A very effective measure aiming at protecting the containment against the late stage of the accident (and the relating issues such as recovery of the source of hydrogen, melt-through of the door or the cavity) would retaining the melted core inside the vessel by means of reflooding the reactor cavity.

Consequences of the door melt-through in the cavity can be extenuated in particular by means of hermetization of the A0065 room or by sealing the interspace between the steel door and the concrete blocks. Cooling debris in the cavity with water may be effective.

Interaction of the melted core with the concrete containment bottom is a well-known phenomenon with VVER-440/213 relating to the break-through of the melted core out of the cavity wall or even sooner out of the steel door in the cavity wall into the non-hermetic area.

A slower phenomenon is the base plate melt-through. It is thicker than the cavity wall, 3.1 m, and under it, there is soil contributing to the fission products´ filtering. The advance of the melted core is generally faster in radial that in axial direction.

Hydrogen generation in this stage of the accident, compared to the hydrogen generation during the steam-zirconium reaction in the early stage of the accident, is considerably slower.

According to the analyses, the side melt-through of debris out of the cavity´s wall would occur in approx. 4.5 days as of the radiation event provided that the two steel doors are not melted through sooner. Water supplied to the cavity after the break of debris through the vessel could extend this period and in particular could protect the steel door.

The strategy of the melted core cooling in included in the SAMG, “Reactor cavity reflooding”. Supply of water to the cavity prior to the vessel´s melt-through is possible at present by means of overflow via the reactor´s footstall only. This would require relatively extensive amount of water, reserves of the ECCS tanks and water from the bubbler trays. This intervention can extend the time until the break of debris through the cavity by several up to many hours. Cooling debris, however, does not have to last until depletion of water; a configuration, which cannot be cooled down completely, may develop due to extensive volume of debris on a small area of the reactor cavity bottom. The SAMG manual assumes discharge of the bubbler trays or pumping water from the ECCS tanks if the spray pumps are operable. The manual considers using the water reserves from the twin unit as well.

After modifications according to chapter 6.3.5.1, cooling debris in the cavity would be useless as the debris would be retained in the vessel. If, after all, the vessel´s bottom is damaged, the cavity would have been reflooded and the steel door would have been protected automatically and the interaction of debris with concrete postponed.

The main benefit of the strategy based on reflooding of debris in the cavity is the steel door cooling. With larger quantities of debris, water is not likely to be able to stop the melted core´s break through the cavity´s concrete.

6.3.5.3 Cliff edge effects related to time delay between reactor shutdown and core meltdown

As it follows from the analyses of the SBO scenario, under which a loss of heat removal from I.O occurs on the part of SG, even without implementation of alternative activities described in EOPs, there is still a relatively long time allowance for the recovery of heat removal from I.O. The temperature of 550 °C at the reactor core outlet was reached in approx. 9 hours as of the SBO occurrence. Similar time allowances were discovered with the “transient” scenario (full loss of the SG feed water supply) as well. If alternative make-up of SG is performed in accordance with EOPs, it is possible to extend this period in the order of days.

The LOCA accidents with a loss of all active systems of safety injection of primary coolant could lead theoretically to sooner damage to reactor core. An example of such accidents is the SBO+LOCA combination. The PSA results indicate, however, extremely low frequencies of such events, lower than 10^{-8}/year. The severe accident analyses thus focus on the more likely scenarios LOCA when the loss of cooling occurs no sooner than in the recirculation stage of the ECCS pumps operation (transfer to suction from the containment). In most cases, we can rely upon a delay in damage to reactor core due to discharging the bubbler trays so that reactor core would get damaged later than in case on SBO in case of failure of the alternative methods of the SG make-up.

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 containments integrity is secured by the following systems:

• The containment insulation system - pneumatically operated QOV and electrically operated QOV on the ventilation systems - the operability conditioned by the ZN power supply of category I.

• The hydrogen disposal system for design basis events contains passive autocatalytic recombination devices (even in case of completion with more efficient ones) and does not require power supply.

• The passive part of the vacuum-bubbler system contains automatic valves which do not require power supply.

• The active part of the vacuum-bubbler system - the active spray system requiring ZN category II - even a long-term failure of which, however, does not result in the loss of the containment´s integrity.

Provision of the necessary power supplies for the functionality of the aforementioned systems is described in chapter 1.1.2.

6.3.6.2 Operational provisions

Most of the strategies described in SAMG (water make-up in the containment, heat removal, maintaining pressure in the containment) require power supply for their successful implementation.

6.3.7 Measuring and control instrumentation needed for protecting containment integrity

The EDU units are equipped with the hydrogen concentration measurement system. The hydrogen concentration measurement system includes 15 detectors with the scope within 10 % deployed in various rooms of the containment.

In case of severe accidents, certain damage to some measuring detectors due to hydrogen fire may occur. Nevertheless, a larger portion of the instrumentation necessary for the SAMG implementation should be available.

The 10% limit of the hydrogen concentration measurement could be exceeded in case of hydrogen fire. The limited scope of the hydrogen concentration measurement does not represent a major problem as the transfer from the strategy based on the hydrogen ignition to the strategy based on the combustion elimination should be realized at the hydrogen concentrations which are lower than 10 %. A deficiency relation to the hydrogen issue management is the absence of the water steam measurement concentration in the containment. For this reason, SAMG contain computational methods providing for the missing information.

For the analysis of the reactor core cooling loss, measurement of temperature at the outlet of reactor core is used. As described in chapter 6.1.3.8, it is necessary to take into account a gradual loss of thermocouples at the reactor core outlet. This would, however, concern the thermocouples; for the strategy implementation, level gauges and pressure measurement are more important. With the exception of level measurement in the cavity as such they are likely to be available and undamaged. A part of them is outside the affected containment - the ECCS tanks and all instrumentation in the twin unit.

Measurements of pressure, temperature and water level on the containment´s floor are functioning under normal operation of the unit and are not designed particularly for emergency conditions. However, their partial availability even under conditions of a severe accident is assumed. For the containment integrity protection, measurement of pressure is decisive as based on it the decision to perform either filtered or non-filtered ventilation of the containment to protect its integrity at the increase of pressure above 300 kPa is taken (if such overpressure is reached at all). Both the measurement and the respective intervention require the ZN I. functionality.

For the assessment of the containment´s threat due to overpressure, there are measurements of overpressure in the containment with sufficient scope (450 kPa) deployed in multiple places. It is assumed that at least one of the measurements could withstand the course of a severe accident.

For the identification of interaction between the melted core and the containment´s bottom´s concrete, there are no direct measurements, however, it is possible to identify it from measurements described below and to estimate it based on the computational method described in SAMG, however, the stated indicator are not as accurate as direct measurement. Interaction of the melted core and the containment´s bottom´s concrete can be identified based on indirect measurements which are available in PAMS. The respective measurement are the ones of pressure, temperature and level in the SG boxes and in particular in the cavity and the reactor vessel, on condition that this previous measurement has withstood the severe accident conditions. Indication of pressure in the primary system, which will decrease to the pressure in the containment (unless it decreased sooner already), is even better. The beginning of the new increase of the hydrogen concentration in the containment could be another indicator, however, the concentration of hydrogen in the containment would exceed the measuring scope of 10 % or would be very inaccurate due to lack of oxygen.

The basic measurement in case of extensive leak of fission products are measurements of dose rates and activity. For measurement of dose rates and activity, it is possible to use the measurements inside and outside the containment, in the stacks and the measurements from the teledosimetric system TDS located on the fence of the plant. The scopes of all these measurements are designed for both operating and accident and post-accident conditions.

The radiation measurements with scope for accident and post-accident conditions are realized in the CISRK system and have not been implemented in PAMS yet.

6.3.8 Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

The EDU units are structurally connected to create twin-units, however, from the point of view of technology, the units are independent of each other. In addition, the twin-unit arrangement enables, in case of necessity, to use media from the twin unit through the common auxiliary systems. Activities concerning the accident management on the NPP individual units are controlled from ECC (TSC and EC) and interventions on individual units are performed by the operating staff of the respective unit. Depending on the actual situation on the individual units, it is possible to move capacities from one unit to the other on an operational basis. In case an accident occurs on one unit, the TSC staff has a manual for decision-making concerning the method of operation and performance of necessary activities on the twin-unit at their disposal. In case, the event develops into a severe accident affecting multiple units, identical SAMG manuals for all the units would be used, nevertheless, the situation on the individual units would be analysed individually and TSC and EC would perform the necessary coordination of activities on the individual units.

6.3.9 Conclusion on the adequacy of severe accident management systems for protection of containment integrity

For the management of beyond design basis and severe accidents, all available technical equipment is always used, even the one which is not primarily designed for the severe accident management. The use of this equipment is described in the respective strategies included in EOPs and SAMG. The strategies focus on success, which means that one of the alternate goals of SAMG is recovery of the systems´ and equipment´s availability within the most extensive scope and the implementation of the given strategy leads to success in any of the described ways. Here success means fulfilment of the principal SAMG goals, which means putting the unit into a stable controlled state and limiting radioactive media leakage.

Although there are several alternative systems for implementation of each strategy, within the framework of the assessment of containment integrity protection measures, 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 RPV melt-through followed by melt-through of the containment´s bottom.

Some measures to manage accidents may be very risky (threat to persons, large leaks of radioactive media, NPP destruction, etc.). Such risks shall be considered carefully beforehand and the staff members responsible for the management of emergency conditions must be able to decide even on taking such risky measures.

From the point of view of severe accidents management, the long-term post-accident activities mean continuance of activities to provide for heat removal and elimination of occurrence of highly-energetic effects (hydrogen combustion or detonation, etc.) depending on the unit condition. In this case, it may be very difficult to define exactly what the condition of the unit is like and thus to define possible threats. Nevertheless, after putting the unit into a controlled stable state, the basic prerequisite for the SAMG termination has been fulfilled. Prior to leaving SAMG and continuing the long-term post-accident activities, SAMG contains descriptions of the ways enabling to identify the unit´s condition in the most accurate manner, to determine the scope of damage and long-term risks.

The long-term post-accident activities are moved from the stage of searching for the suitable measure to the stage of provision of a long-term functionality of the found and applied successful measures, which means e.g. provision that no interruption of the alternative water supply will not occur due to any reason (loss of power supply, exhaust of water reserve, failure of components). This also concerns searching for alternatives to the already successfully implemented measures, i.e. searching for other measures, which after the provision thereof, immediately replace the already applied ones or the stand-by ones in case of loss of the currently implemented measures.

6.3.10 Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

To improve the ability to maintain the containment integrity after a severe damage to fuel, it is possible to propose and implement the means for the provision of the containment integrity provision (which means eliminating fission products leakage) during a severe accident (hydrogen disposal, melted core confinement).

The opportunities for improvement of the in-depth defence during events, which may result in occurrence of severe accident, are provided in the following table. All the mentioned provisions (designated as "PSR finding" in the note) would be implemented even without this goal-directed review, which has confirmed in its outcomes the effectiveness and correctness of the previously taken decisions to implement provisions aiming at upgrade of the original design. A more detailed description of the corrective actions is provided in the annex.

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
Containments´s integrity during severe accident	Increase of capacity of the system of recombination of hydrogen produced due to accident	II	PSR finding
Localization of the AZ meltdown	Cooling meltdown from outside of TNR	II	PSR finding

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 leaks of fission products from the plant after the reactor core damage may represent a threat to health and safety of persons. Extensive leaks (General emergency according to the On-site emergency plan) are defined as leaks exceeding the criterion for imposition of radiation MU-3 according to the Emergency plan (100 microSv/h).

Depending on the behaviour of the accident, radioactivity may be released as follows:

- Directly into the containment and after the loss of its integrity to the environment.
- Through SG, to the secondary system and the environment.
- To the non-hermetic rooms.
- To the ESW system.

Upon assessing leaks of fission products, the biological effects of ionizing radiation, which have two aspects - stochastic and deterministic, have been taken into account. From the point of view of stochastic effects, for which there is no threshold value, it is not possible to set any limiting conditions. From the point of view of deterministic effects, determination of limiting conditions needs to be based on the applicable legislation and normative documentation. Determination of limiting conditions from the point of view of biological effects of fission products exceeds the scope of this assessment.

6.4.1.2 Operational provisions

The strategies are described in SAMG, the manuals titled “Limitation of fission products leakage” and “Extenuation of fission products´ effects”. The individual activities to limit fission products´ leakage differ depending on the route along which fission products are released. In case of leak to the containment and via the damaged or non-hermetic containment into the environment, the primary method is to reduce the use of the spray system, which method is very effective, in case of its unavailability, the use of ventilation systems equipped with heat removal.

As a primary and a very effective measure in case of leak via SG, closure of the both Main Loop Isolation Valves on the damaged loop is applied; the other measures mentioned in SAMG are realizable on the SG steam piping.

From the point of view of the protection of the staff and other persons, RMMS has been established within the framework of POHO, which monitors and analyses radiation situation in the affected areas. For the needs of the preventive measures imposed to the purpose of the persons´ protection, a SW means titled RTARC is available - see chapter6.1.2.3

6.4.2 Accident management after uncovering of the top of fuel in the fuel pool

The fuel storage pools are located in the reactor hall shared by two units. The analysis of the accident behaviour in the fuel storage pool for shutdown states is planned to be performed in 2012. Behaviour to be analysed in the pool under is mode 6, which means during refuelling, mode 7 upon complete removal of fuel from the reactor and in modes 1 through 5, when the storage pool together with the reactor hall is hermetically separated from the containment.

Cooling of the spent fuel storage pool (SFP) is realized by means of two cooling circuits. Each of the cooling circuits includes a circulation pump a heat exchanger. The heat exchangers are cooled by essential service water (ESW 1 and ESW 3). The SFP cooling pumps as well as the ESW pumps are power supplied from DG.

For alternative make-up and heat removal from SFP in case of full loss of the SFP normal cooling (either due to drop of level, loss of coolant or interruption of heat removal), the strategy based on the use of the accumulation functions of the ECCS systems´ tanks had been prepared. In addition, the possibility of SFP make-up with coolant from the twin unit, or possibly by means of the FB equipment is being considered. However, no detailed procedures have been prepared in relation to the aforementioned so far.

The issue of the SFP cooling, or the loss of coolant from the TG cooling circuit, is resolved within the EOPs. The residual output of fuel can be bled to the ECCS tank for a limited time. The accumulation capacity of the full ECCS tanks is approx. 4 days.

The deep sub-criticality of the spent fuel in the storage pool is guaranteed on the one hand by the coolant, the boron concentration of which is 12 g/kg, and on the other hand, by the use of borated steel in the construction of the storage racks. The use of borated steel as such guarantees sub-criticality even in case when the spent fuel is cooled by pure water. In addition, in case of boiling, only water would evaporate (it is assumed only minor priming of boron by leaking steam) and thus the possible make-up with pure water will not lead to a significant decrease of boric acid concentration in the coolant.

As an alternative, make-up of coolant from the bubbler trays, coolant from the twin unit and flooding SFP from the reactor hall by means of the FB equipment is assumed. The respective procedures, however, have not been prepared yet.

After interruption of heat removal from SFP, a permanent increase of temperature would occur, which would be important in particular in case of fulfilment of the upper grate. Without recovery of heat removal, uncovering of fuel would first occur in the upper layer with the follow-up risk of damage to the cladding and fuel melting in the early stage of a severe accident.

With respect to the fact that the storage pools are not located in hermetically separable rooms (only the enclosure of the reactor building), leakage of radioactive media to the NPP environment would follow. In case of occurrence of the steam-zirconium reaction, there would be leak of hydrogen to the space of the reactor hall.

With respect to the existence of the alternative method of heat removal using its accumulation in the ECCS tanks, a long-term loss of heat removal from SFP is not expected as from the point of view of the time necessary for the performance of activities aiming at recovering the cooling of spent fuel deposited in SFP, the situation is more favourable than in case of loss of heat removal from reactor core. Therefore, no analyses have been performed with respect to damage of the spent fuel deposited in SFP.

Under the current stage of the design, no alternative systems for cooling, or coolant make-up to SFP are available.

6.4.2.1 Hydrogen management

The risk of hydrogen presence in the reactor hall, which means outside the containment, has been assessed and as it follows from the results, even a failure of cooling in both the pools is not likely to lead to such a concentration of hydrogen in the reactor hall, which could reach the limit of hydrogen ignition.

Accumulation of hydrogen in other spaces is theoretically possible, however, has not been quantified in detail as a large portion of the hydrogen would burn directly in the containment.

By spontaneous or controlled combustion or recombination of hydrogen, the risk of its accumulation outside the containment decreases. After consumption of all oxygen, accumulation of hydrogen may occur in a part of the containment, as described in chapter 6.2.3

Leakage of the accumulated hydrogen from the containment to other rooms or buildings may occur in the following manner:

a) natural leaks of the containment,
b) in case of insufficient insulation of the containment,
c) by means of purposeful ventilation of the containment.

Ad a) The issue of hydrogen accumulation outside containment due to its natural imperfect leakproofness has not been resolved in detail so far as it is insignificant compared to the accumulation of hydrogen inside the containment and the relating threat of damage to the containment by hydrogen explosion as it follows from the hydrogen-related risk analyses.

Ad b) Most of the containment isolation valves (QOV) is pneumatically operated by high pressure air (with the pressure tank sufficient for several cycles), the control id secured from ZN of category I (an accumulator battery). The valves are qualified for accident conditions and therefore, no failure of closure thereof should occur in the early stage of the accident. An exception might be a long-term state of SBO when a loss of ZN I after discharge of the battery would occur.

Ad c) This case is considered in practice neither in this stage nor in the following ones.

The risk of hydrogen build-up outside the containment due to the aforementioned reasons is insignificant.

6.4.2.2 Providing adequate shielding against radiation

Due to even partial damage of fuel, contamination of the reactor hall of the twin units by radioactive media released from the part-melted fuel located in SFP would occur. In this case, all the staff would have to leave the reactor hall immediately. Under modes 6 and 7, the containment can be interconnected with the reactor hall and therefore, the concurrent contamination and unavailability of the containment could occur.

6.4.2.3 Restricting releases after severe damage of spent fuel in the fuel storage pools

The procedures to resolve the accident including fuel melting in SFP have not been prepared yet. Although the MCR and TSC staff does not have at its disposal the so-called Shutdown SAMG (SAMG for shutdown states). Nevertheless, the available possibilities are well-known and lie in continuation of water make-up and heat removal and possible insulation of leakage from SFP according to the EOPs procedures. Damage would occur after a relatively long time with the exception of mode 7, which provides for sufficient time for an operational solution.

The principle solution for elimination of releases to the environment is stopping or deceleration of the accident by reflooding SFP with water. An emergency system of the pools reflooding is under preparation and the system will be linked to other measures taken in the reactor hall, which exclude the presence of the attending staff.

The reactor hall has a large volume, which has a positive impact on dilution of fission products. The other measures to be possibly taken to prevent release are as follows:

In case of leakage of radioactivity from SFP (or from the reactor under mode 6) is to switch off the high-capacity ventilation systems of the reactor hall immediately; this procedure is already implemented in the existing EOP for shutdown states.

After all the staff have left the reactor hall, it is important to eliminate all possibilities of the staff´s access to the reactor hall.

In case the unit is under mode 6 or 7, which means during refuelling or full removal of fuel from the reactor when the containment is usually interconnected with the reactor hall by multiple penetrations, it is necessary to switch off the ventilation systems of the containment, to provide for departure of all persons from the containment and to close all access routes to the containment of the unit under modes 6 or 7. These measures are induced by the fact that it is not possible to separate the containment from the reactor hall quickly.

6.4.2.4 Instrumentation needed to monitor the spent fuel state and to manage the accident

Measurements characterizing the SFP condition (temperature, level, TG flow rate) are available on the MCR panels only. Measurements of parameters relating to the SFP cooling is not led-out to ECR, nor they are available in PAMS. Similarly, measurements of radiation situation in the reactor hall near the SFP are not available in PAMS.

With respect to the large volume of the reactor hall, its worse leakproofness and low residual output of the fuel, the conditions are not expected to be so unfavourable as inside the containment. Thus most measurements will remain unavailable. In addition, the most important measurements are the one of activity in the atmosphere and the ones of water level in SFP. Availability of the measurements is discussed in a greater detail in chapter 6.4.1.2.

6.4.2.5 Availability and habitability of the control room

The MCR and ECR rooms are located in rooms adjacent to the containment. This section might be exposed to radiation at higher pressures and at high doses inside the containment occurring at the same time or in case of extensive leaks of fission products from the containment. Equipping the MCR and ECR with the filtration ventilation systems has not been completed yet. Execution of interventions according to EOPs, or according to instructions issued by TSC upon transfer to SAMG assumes preservation of the MCR habitability. In substantiated cases, shift supervisor or safety engineer or unit supervisor may decide on the transfer of the MCR staff to ECR. Usage of breathing apparatuses in MCR falls within the competence of unit supervisor.

Prior to completion of making MCR and ECR more resistant, evacuation of the staff upon the EC commander´s order based on the assessment of radiation situation is assumed while adhering to the criteria imposed by the intervention instruction (subsequently, only short-term presence to perform interventions would be assumed). The issue of the permanent MCR habitability is discussed in chapter 6.1.3.4.

A direct impact of accidents in the containment on habitability of MCR of the unit affected by break through the containment wall to MCR and ECR will be subject to additional assessment.

In the early stage of the loss of the SFP cooling until the beginning of boiling and decrease of level in SFP, access to the reactor hall is possible. Due to the decrease of level in the storage pool and coolant boiling, an increase of the dose rate in the area surrounding the pool would occur and its environment would become inaccessible. The activity would gradually spread around the entire hall and the space of the twin unit as well and all the staff would have to leave the reactor hall.

Due to even partially damaged fuel contamination of the reactor hall of both the twin units by activity and the unavailability thereof would occur. Under modes 6 and 7, the containment can be interconnected with the reactor hall and therefore, the concurrent contamination and unavailability of the containment could occur. Habitability of all MCRs could be indirectly affected due to leakage from the reactor hall to the environment and due to the radioactivity suction by the ventilation system or by means of radiation from the space before MCR.

6.4.3 Conclusion on the adequacy of measures to restrict the radioactive releases

Although the elimination of a loss of the containment integrity as the last barrier against the leakage of fission products to the environment together with the limitation of release of fission products are principal goals in SAMG, the SAMG also contains descriptions of strategies aiming at terminating or at least reducing releases of fission products after the loss of the containment integrity using all available means.

The respective manuals for orderly use of all the available possibilities to eliminate leakage from the reactor hall will be prepared.

6.4.4 Measures which can be envisaged to enhance capability to restrict radioactive releases

Although there are several diverse systems to implement each of the accident management strategies, there exist opportunities for further safety enhancement in the severe accident management system. 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 basis SSKs is concerned. Upgrade of the EDU´s design in the area of severe accident consequences was decided on as early as within the framework of PSR. In the area of technical solutions, increase of capacity of the existing system of recombination of hydrogen produced due to accident is being prepared. They are related to the fact that given the existing design capabilities, possible containment integrity challenge by hydrogen produced in severe accidents cannot be fully eliminated. Also, the provisions to ensure the possibility of external flooding of the reactor pressure vessel are not finalized.

In view of the administrative management, the guidelines for severe accident management for shutdown conditions and the cases of fuel failure in the spent fuel storage pool have not yet been completed..

The opportunities for improvement of the in-depth defence during events, which may result in occurrence of severe accident, are provided in the following table. All the mentioned provisions (designated as “PSR finding” in the note) would be implemented even without this goal-directed review, which has confirmed in its outcomes the effectiveness and correctness of the previously taken decisions to implement provisions aiming at upgrade of the original design. A more detailed description of the corrective actions is provided in the annex.

Opportunity for improvement	Corrective action	Deadline (short-term I / medium-term II)	Note
Containment’s integrity during severe accident	Increase of capacity of the system of recombination of hydrogen produced due to accident	II	PSR finding
Localization of the AZ meltdown	Cooling meltdown from outside of TNR	II	PSR finding
Regulations	Prepare “shutdown SAMG” for outage / severe accident in BSVP	I	PSR finding

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 EDU 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 defence in depth and the capability to fulfil the fundamental safety functions during the specific initiating events and even severe accidents regardless of extremely low probability of their occurrence. The evaluation was performed for all operating modes and unit reactor states, including the case if all EDU 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 EDU shall not exceed, with a 95 % probability, 6°MSK-64 (PGAupper = 0.06 g). The actu al resistance of buildings, systems and structures is higher, thus safety margin is available for the remaining 5 % uncertainty.

At present, the safety related equipment and civil structures are being reinforced in all units to withstand the peak ground acceleration value PGA = 0.1g (maximum design earthquake, MDE/SL2/SSE). Over 90% (amongst others, all the process) of the components and structures important to safety currently have satisfactory qualification documentation to prove their seismic resistance and modifications are being finalized and implemented for the rest of components and structures (electrical and I&C).

7.1.2 Robustness against floods

The EDU site is not endangered by natural flooding. The power plant site is located on a plateau at 383.5 – 389.10 m above sea level and the main civil buildings in which safety related components and structures are installed lie at the upper edge of this interval. The closest water stream is the river Jihlava, which is used to supply make-up service water to the plant. The system of waterworks Dalešice – Mohelno on the river Jihlava cannot jeopardize safety of EDU in extreme floods, nor in case of dam failure of both waterworks.

The waterworks Dalešice located up the river (distance of app. 4 km from the power plant, the dam height 88 m) with crest level 384.00 m n. m. above sea level and the maximum water level (at spillway due to floods) at the elevation of 381.50 m above sea level. The waterworks Mohelno is located about 2 km down the river and its dam crest is at 307.15 m above sea level and the maximum water level (at spillway due to floods) at the elevation 303.30 m above sea level, i.e. by app. 80 m lower than the civil buildings of EDU.

The annual course of precipitation is characterized for a long term by the highest precipitation amount in summer season, the maximum in June (70 mm), and the lowest amount in winter season, the minimum in January (21 mm). The drainage network is designed as a branch structured system, ensuring rainwater gravity drain from the area of app. 80 ha and connecting to the final rainwater sewer in front of the EDU site.

The actual one-day storm rainfalls correspond to a created level of 77 mm on the EDU site (100-years precipitation amount). The civil structures of EDU are designed to withstand floods up to the maximum level of 115 mm (10,000-year maximum 24 hour precipitation). The difference of the above values provides roughly a 30% safety margin. The essential design provisions against flooding of safety related structures and components by rainfall is the siting of the power plant with gravity rainwater drainage from the area of app. 80 ha, and sufficiently rated storm-water drainage system, and laid out elevation of entrances and gateways with respect to the surrounding area and sloping of the adjacent roads.

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 weather effect parameters for the EDU site are based on statistical processing of annual extremes of the relevant meteorological parameters for at least 30 years of measurement of such events on the EDU site, or at meteorological stations in the surrounding region.

The following table shows the specific derived extremes of weather effects on the EDU site, including the relevant design values and extreme loads of buildings (except of precipitation:

Event (Weather Effect) / Parameter	**100-year recurrence**		**10,000-year recurrence**	
	Value	**Load**		**Value**
Gusty wind / speed	46.2 m/s	0.69 kN/m^2	60.6 m/s	1.26 kN/m^2
Snow / water column equivalent	109.0 mm	1.09 kN/m^2	195.0 mm	1.95 kN/m^2
Maximum temperature / absolute maximum per year	39.0 °C	-	46.2 °C	-
Minimum temperature / absolute minimum per year	- 30.8 °C	-	- 46.7 °C	-

Some partial differences in actual resistance of selected buildings from the required values of resistance under extreme load (which, however, cannot compromise the fulfilment of safety functions) are addressed in the currently finalized project of supplemented seismic qualification of the components and civil structures important to safety.

7.1.4 Robustness against loss of power

The power supply sources at EDU 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, mutual backup and redundancy (see working and standby house load sources as well as emergency sources of alternate and direct-current power supply, so called essential power supply systems that feed safety related systems and components).

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

There are a total of 12 emergency alternate power sources available on the site (DGs). In the mode of loss of off-site power, the reactors at EDU can be maintained in fail-safe state for a long term, 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 6 to 7 days, without a need for external refuelling.

In case of complete loss of the AC power supply (SBO), the only power supply for the safety systems and the safety-related systems are the DC uninterrupted emergency power supply systems (accumulator batteries). Without the respective DG operation, the accumulator batteries are not recharged and the time available prior to their discharge is in the order of hours up to tens of hours depending on the currently applied load. This time is sufficient for recovery of power supply of the EDU´s VS units from the neighbouring Dalešice or Vranov hydraulic power plants.

7.1.5 Robustness against loss of ultimate heat sink (UHS):

Ultimate heat sink for reactors at EDU 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 ESW (essential service water) system.

On-site water inventory can last for about 39 days of operation of ESW system, ensuring decay heat removal from shutdown reactors of EDU without external water replenishment to ESW system. A total of 12 ESW pumps are available for each unit (2 reactors). Loss of all ESW pumps could be caused by parallel loss of electrical power to both reactors per each unit.

The robustness of EDU in case of possible loss of all ESW corresponds to the scenario after the occurrence of SBO. If the loss of ESW was not combined with SBO, an alternative method can be used for heat accumulation from spent fuel storage pools (SFP) to ECCS, or SFP evaporated coolant replenishment from bubbler trays. Accumulation capabilities of ECCS tanks cover more than 4 days, possibility to make up water in SFP from bubbler trays is app. 13 days. Alternative possibility is using Fire Brigade to keep level of boiling water in SFP.

7.1.6 Severe accident management system robustness:

EDU has implemented a severe accident management system to ensure level 4 defence in depth and an emergency preparedness system to ensure level 5 defence in depth. The functioning and interconnected accident management and emergency preparedness system at EDU 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 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 EDU with the assistance of the technical support centre.

In case of an abnormal event, all the necessary activities would be managed and performed from protected places. 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 even in case of activity leak into the environment. Remote activities to implement strategies would be performed by the shift personnel from MCR or ECR where the occupancy project for the control centers is being finalized. Local activities and possible equipment repairs in the relevant areas of the reactor hall, turbine hall or external buildings would be provided by intervention teams located in the operational support centre.

The concept of the accident management system at EDU 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 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 unfailed 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 the event 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 organisation of abnormal event management (including severe accidents) is defined in the on-site emergency plan approved by SUJB (State Office for Nuclear Safety).

In 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 at the EDU 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 EDU 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 specific scenarios of severe accidents.

7.2 Safety issues

The assessment of safety margins of EDU 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 defence 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 EDU 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 fulfil 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 defence 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 fulfil the required functions, or take consequent protective actions to reduce personal exposure for protection of people.

Annex

Description of the proposed corrective actions

The Annex contains a brief description of the proposed corrective actions to be taken in all areas subject to review within the framework of the Stress tests. For each corrective action, the following details are provided:

Position

Safety function covered by the respective corrective action.

Method of solution

A brief description of the method of solution with specification of the proposed available alternatives.

Risks

The risks, which are solved by means of implementation of the proposed corrective action.

Next step

A brief proposal of the method of implementation of the proposed corrective action

<table>
<tr><th></th><th>POSITION</th><th>NO - METHOD OF SOLUTION</th><th>RISKS</th><th>NEXT STEP</th></tr>
<tr><td rowspan="5">ALTERNATIVE TECHNICAL MEANS</td><td>1.Heat removal from AZ through II.O</td><td><u>Water volume control in PG in an alternative manner from the tanks of demineralized water 1 MPa or from the off-site source of supply:</u>
(i) Installation of stable motor pump (diverse SHNČ)
(capacity approx. 160 kW)
(ii) PG´s volume control by means of the fire-suppression equipment
(necessary to take into account the capacity of HZSp)</td><td>UHS
SBO</td><td rowspan="5">These concern diverse (independent) means connected to the electrical distribution systems / pipelines of the existing BSs, which means a solution affecting the existing BSs.
IMMEDIATELY: coordination with the other operators of VVER / WANO / VVER designer – specification of the solution concept
2012: feasibility study</td></tr>
<tr><td>2.Heat removal from AZ through I.O.</td><td rowspan="2"><u>Water volume control in I.O/BSVP in an alternative</u>:
(i) Filling the opened reactor and BSVP by gravity from the XL conduits
(analyse the necessity of reconnection of the valves to the ZN I system)
(ii) Filling the closed reactor at low pressure using the XL10D01 pump with a higher capacity and a new pipeline from the conduits to the pipeline from HA to Re
(power supply for the pumps and the valves from ZN1, the proposed solution must be analyzed thoroughly)
(iii) BSVP´s cooling by means of the coolant volume control and heat accumulation in the TH tanks</td><td>UHS
SBO</td></tr>
<tr><td>3.Pumping media for BSVP (including the BSVP´s filling)</td><td>Floods
UHS
SBO</td></tr>
<tr><td>4.Provision of power supply of category I</td><td rowspan="2">One mobile DG with a capacity of 500 kW per unit (4 DGs per power plant) for ZN of category I + voltage for the selected current consumers 0.4 kV of category II
It is not designed for power supply of pumps with high capacities, however, enables to power supply gradually the selected drives of 0.4 kV ZNII - VZT BD and ND, TM pumps, TH or TQ (not at the same time).
Connection by means of three cables to EV, EW, EX, which enables to power supply ZN1 (+ recharging batteries) and partial power supply of ZNII.
Note concerning the cooling of SKŘ: analyse heat removal under the SBO conditions, the conclusion may be to disconnect the selected boxes, to open the doors; it is likely that no special means will be necessary</td><td>SBO</td></tr>
<tr><td>5.Habitability of BD/ND</td><td>SBO</td></tr>
</table>

	6. Provision of heat removal from the SKŘ´s means		SBO UHS	

	POSITION	NO - METHOD OF SOLUTION	RISKS	NEXT STEP
REGULATIONS	Completion of PP and HPP (off-site events, loss of UHS, power supply recovery)	Prepare PP to cover earthquake, extreme events (wind, temperature, snow), power supply recovery after SBO of all units, loss of UHS, volume control of PGs of all the four units by means of the fire-suppression equipment	Seismicity, extreme weather, UHs	2012: Complete PP, implement provisions
	Extension of the discharge time of the accumulator batteries of ZN cat. I during SBO	Analysis of the discharge time of the accumulator batteries while implementing the variants of the controlled reduction of consumption (disconnection of current consumers). Completion of PP for the SBO´s mode with instructions for reduction of load of ZN cat. I. Change of connection and operation of the emergency lighting system (project 5663).	SBO	
	SAMG for outage / severe accident in BSVP	Prepare the "Shutdown SAMG".	AMP	2012: Commence
	Guidelines / EDMG for alternative means	Procedures for handling and use of the alternative means for emergency cooling or emergency power supply	Seismicity Floods Extreme weather SBO, UHS, AMP	In relation to the solution of the alternative technical means

	POSITION	NO - METHOD OF SOLUTION	RISKS	NEXT STEP

	Position	NO - Method of solution	Risks	Next step
Personnel, Communication	Means to support actions taken by the personnel - power supply for the means of communication, lighting, revolving doors	<u>Communication between the intervening persons and off-site</u> (a) Within the premises of the power plant (BD – TPS – intervening personnel …) i) Alternative power supply of the telephone exchanges - approx. 20 kW at AB1 - approx. 10 kW Cross intermediate room 3rd RB **At AB1, supply power from the standby power supply unit for the AB1 shelter.** **At the 3rd RB, power supply from the emergency power supply unit for the PB2 shelter.** The same power supply units shall be used for emergency recharge of portable torches and freesets (solve by means of installation of 2 recharging points - e.g. shelters AB1 and PB2). (Note: the existing back-up of the exchanges upon failure of power supply is from own accumulator batteries for a period of approx. 8 hours) ii) Determine the concept for communication upon disintegration of the communication network due to seismic event *(all the existing communication infrastructure(telephones, radio stations) is dependent on decentralized elements located in non-seismic civil structures, non-seismic cable ducts, etc. – at the present time, we have redundant communication possibilities (fixed lines, MT, radio stations), however, without guarantee of full functionality after PZ/MVZ.* *It is proposed to use* independent radio stations or stations functioning through mobile converters (in vehicles of HZSP and radiation protection) …to be solved in cooperation with the other operators. (b) Between the power plant and the off-site authorities See ii) above. Verify the possibility of communication of the key workplaces (power stations, authorities) by means of satellite telephones. <u>Power supply of TSFO</u> **Power supply of TSFO is to be solved by means of two smaller emergency power supply units (DGs) with total capacity of approx. 135 kW per power plant.** The proposal of two smaller DGs is based on the concept of power supply of TSFO from two switchyards 9BA, 9BB. The capacity of DGs necessary for TSFO needs to be specified in relation to the planned refurbishment of TSFO (Note: the existing back-up of the revolving doors upon failure of power supply is from own accumulator batteries for a period of approx. 4 hours)	Seismicity SBO	2012: Coordination with the authorities and IZS with respect to the possibility of alternative warning, reporting and communication in case of damage to the existing infrastructure. Modification of the emergency planning documentation. Coordination with the other operators in the area of the communication´s solution. Possible analysis of the elements´ resistance during PZ - prerequisite of confirmation of at least limited functionality preserved

	Position	NO - Method of solution	Risks	Next step
	Power supply of the means of warning	Define the alternative organizational solution (in cooperation with the authorities and IZS) in case of failure of the radio and hooters (or of the control infrastructure for the activation thereof) due to extreme natural disasters (equipment with mobile means - mechanic hooters, pneumatic hooters, loudspeakers on vehicles, hooters on vehicles), including the method of use and processing in the emergency planning documentation.	AMP	
	Upgrade and the operability of the shelters	Execute analysis of the shelters´ exposure during seismicity and floods - in particular AB (HŘS) and PB2. **Implement ZKZ 6355 (standby power supply of the shelters).**	Seismicity Floods SBO AMP	2012: Solution / Commence implementation of the provisions.
	Provision of sufficient staff	i) Provision for altering members of the shift staff in case of aggravated availability of the location. ii) Analysis of possibilities of the shift staff in case of occurrence of accident on all the four units (sufficient number of persons to implement the respective strategies, sheltering thereof). iii) (Upon occurrence of EDMG - analyses of conditions and possibilities / sufficient staff for implementation of actions based on EDMG.) *Note: Prompt evacuation of persons from EDU (who are not participating in mitigation of the accident) in case the shelters are inoperable - HP´s provision.*	Seismicity Extreme weather SBO AMP	
	Access to interventions	Verify accessibility of heavy equipment. The HŘS Logistician has a list of equipment at his disposal - contact the Air Force Base Náměšť nad Oslavou Analysis of access for the HZSp´s equipment to HVB I, II	Seismicity	
	Usability of IHŘS - ability of the OHO´s functioning outside HŘS	Activation of TPS and HŠ outside HŘS (including handover of information and communication) - verify possibilities, incorporate in the documentation	Seismicity SBO AMP	
	Training for severe accidents	Organizational provisions - TPS training	AMP	
	Cooperation with the off-site services	Prepare agreement with off-site services (IZS, Armed Forces of the ČR) and near NPPs. Organizational provision, to be solved within the framework of HP.	AMP	

	POSITION	NO - METHOD OF SOLUTION	RISKS	NEXT STEP
TECHNICAL MEANS - DESIGN BASIS (BN, PSR)	Complete the design of seismic upgrade, including anchoring of the non-seismic equipment	ZKZ 5239, 5321, ZKZ from the area of technological equipment	Seismicity	To be solved within the framework of BN, PSR
	Diverse CHV	ZKZ 5983	Extreme wind, UHS	
	Modification of the PV PG´s control	ZKZ 4563	UHS, SBO	
	Provision of habitability of BD	ZKZ 3706	AMP	
	Oxygen regeneration in the shelters	ZKZ 6301	AMP	
	Recombination of post-accident hydrogen	ZKZ 5192	AMP	
	Cooling meltdown from outside of TNR	ZKZ 5778	AMP	

	POSITION	NO - METHOD OF SOLUTION	RISKS	NEXT STEP
TECHNICAL MEANS (OUTSIDE BN, PSR)	Anchoring of non-seismic equipment (e.g. transport roller in the Re hall)	Visual inspection, ZKZ	Seismicity	IMMEDIATE:
	Resistance of the HZSp´s means	Analysis of resistance of the HZSp´s building to seismicity, wind, snow or the HZSp´s building´s upgrade. ZKZ. Review of capacity of the HSZp´s means, test of power supply for PGs of two units by means of one power supply package.	Seismicity, extreme weather	

	Upgrade of civil structures to withstand extreme wind and snow	Preparation of the methodology for assessment of external influences, verification of the executed analyses, possible technical provisions (ZKZ 6697) Verify the conclusions of the M.L.E.&C analysis, or implement the respective technical provisions respectively	Extreme weather	MEDIUM-TERM: analyses, feasibility studies
	Completion of PAMS with measurements of RA situation and BSVP	RA measurements initiated by ZKZ 5812 (Sejval), TPo 6740 issued (completion of PAMS), 6789 (SYRAD) BSVP - new measurement of level and temperature in BSVP	AMP, SBO	
	Protection of BD against radiation	Verification of the analysis by ÚJV Řež (opponent opinion), possible upgrade	AMP	