National Report on

„Stress Tests“

NPP Dukovany and NPP Temelín
Czech Republic

Evaluation of Safety and Safety Margins in the light of the accident of the NPP Fukushima

State Office for Nuclear Safety
Czech Republic

Revision 1 – March 2012

Contents

LIST OF ABBREVATIONS

Abbreviation	Description
AAC	Additional Alternate Current Source
AC	Alternate Current
AOPs	Abnormal Operating Procedures
ATWS	Anticipated Transient Without Scram
BCEQ	Bubbler Condenser Experimental Qualification
BES	Backup Emergency Switching
BSHC	Backup Supply of House Consumption
CBSS	Cooling Basins with Sprinkler System
CCC	Crisis Coordination Centre
CDF	Core Damage Frequency
CO	Carbon Oxide
CPS	Central Pumping Station
CR	Control Room
CRMS	Central Radiation Monitoring System
CSS	Containment Spray System
CT	Cooling Tower
ČEPS	Czech transmission grid
ČEZ ICTS	CEZ communication network
DBE-1	Design Basis Earthquake
DBE-2	Maximum Design Basis Earthquake
DBF	Design Basis Flooding
DE	Design Earthquake
DG	Diesel Generator
DGS	Diesel Generator Station
DID	Defence in Depth
DIDELSYS	Defence of Depth of Electrical Systems
DSR	Detailed Seismic Regionalization
EC	European Commission
DG SANCO	DG Health and Consumers
ECC	Emergency Control Centre
ECCS	Emergency Core Cooling System
ECURIE	European Community Urgent Radiological Information Exchange system
ECR	Emergency Control Room
E&CR	Emergency & Control Rod
EdF	Electricité de France
EDMG	Extensive Damage Mitigation Guideline
EE	Extraordinary Event
EFWP	Emergency Feed Water Pump
EH	Emergency Headquarters

ELS	Emergency Load Sequencer
ENSREG	European Nuclear Safety Regulators Group
E.ON	Power company
EOPs	Emergency Operating Procedures
EPZ	Emergency Planning Zone
ERB	Emergency Response Board
ERO	Emergency Response Organization
ESFAS	Engineered Safety Features Actuation System
ESW	Essential Service Water
EU	European Union
feed&bleed	Reactor cooling down method
FA	Fuel Assembly
FAV	Fast Acting Valve
FB	Fire Brigade
FWT	Feed Water Tank
HA	Hydro Accumulator
HC	House Consumption
HCLPF	High Confidence on Low Probability Failure
HELB	High Energy Line Break
HPS	Hydropower Station
HZ	Hermetic zone
I.C	Primary circuit
II.C	Secondary circuit
I&C system	Instrumentation and Control system
INSAG	International Safety Advisory Group
IAEA	International Atomic Energy Agency
IOER	Internal Organization of Emergency Response
IPERS	Independent Peer Reviews of Probabilistic Safety Assessment
IRS	Integrated Rescue System
ISFS	Interim Spent Fuel Storage
ISRC	Information System of Radiation Control
LERF	Large Early Release Frequency
LFRU	Local Fire Rescue Unit
LOCA	Loss of Coolant Accident - design basis accident DBA
LOOP	Loss of Offsite Power
MCP	Main Circulating Pump
MCR	Main Control Room
MDE	Maximum Design Earthquake
MPLS WAN	MPLS Wide Area Network
MPU	Main Production Unit
MSIS-64	Macro Seismic Intensity Scale
NFS	New Fuel Storage
NPP	Nuclear Power Plant

NRI	Nuclear Research Institute
NRPI	National Radiation Protection Institute
NSRS	Non Safety Related System
OCS	Operating Control System
OECD	Organisation for Economic Co-operation and Development
OER	Organization of Emergency Response
OIs	Operating Instructions
PAMS	Post Accident Monitoring System
PACHMS	Post Accident Containment Hydrogen Monitoring System
PC	Personal Computer
PGA	Peak Ground Acceleration
PRPS	Primary Reactor Protection System
PRV	Pressurizer Relief Valve
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSV	Pressurizer Safety Valve
RA (Ra)	Radioactive
RC	Reactor Core
RCS	Reactor Control System
RIS	Radiation Instrumentation System
RLS	Reactor Limitation System
RPV	Reactor Pressure Vessel
RTARC	Real Time Accident Release Consequence - computer code
RTS (HO-1)	Reactor Trip System
RUS	Reactor Unit Supervisor
SA	Severe Accident
SAG	Severe Accident Guidance
SAMG	Severe Accident Management Guideline
SAR	Safety Analysis Report
SBO	Station Blackout
SBSA	Steam Bypass Station to Atmosphere
SCG	Severe Challenge Guideline
SDEOPs	Shutdown Emergency Operating Procedures
SE	Shift Engineer
SEFWP	Super Emergency Feed Water Pump
SFS	Spent Fuel Storage
SFSP	Spent Fuel Storage Pool
SG	Steam Generator
SGSV	Steam Generator Safety Valve
SIS	Safety Injection System
SMS	Seismic Monitoring System
SOER	Standby Organization of Emergency Response
SPSS	Secured Power Supply System

SPS I (II)	Secured Power Supply 1st/2nd category
SRS	Safety Related System
SS	Safety System
SSC	Systems, Structures, Components
SÚJB	State Office for Nuclear Safety
TB system	Boric acid make-up system
TC system	Continuous primary coolant purification system
TD system	Teledosimetric system
TE system	Primary coolant drainage system
TG	Turbo Generator
TG system	Spent fuel pool system
TH system	Core emergency cooling system – low pressure part
TH tanks	Tanks of low pressure ECCS (TH system)
TJ system	Core emergency cooling system – high pressure part
TK system	Chemical and volume control system
TL system	Ventilation system
TLHC	Total Loss of House Consumption
TM system	Water treatment of spent fuel storage pool system
TQ system	Spraying system
TSC	Technical Support Centre
TSG	Technical Supporting Group
TSPP	Technical System of Physical Protection
UCTE	Union for the Coordination of the Transmission of Electricity
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
US DOE	United States Department of Energy
VBC	Vacuum Bubbler Condenser
VVER	Pressurized water reactor of Russian provenience
WANO	World Association of Nuclear Operator
WANO MC	Moscow Centre of WANO
WENRA	Western European Nuclear Regulator´s Association
WHO	World Health Organization
XL	Bubbler condenser system

Introduction

The accident at the Fukushima Dai-ichi nuclear power plant in Japan created a need in the EU to evaluate and assess resistance of European nuclear power plants against extreme and very unlikely events for which these power plants might not be sufficiently equipped because such events were not taken into consideration in the original design. The requirement of the European Commission (EC) to perform “Stress tests“ was sent to the EU member countries on 24th May 2011. The aim of the stress tests is to determine the extent of existing safety margins and time intervals after which accidents become “severe accidents” with damaged fuel rods and with extensive radioactive release into the surroundings. Requirements for the technical content of the reports were defined by the European Nuclear Safety Regulators Group - ENSREG. The requirements for the national reports were also elaborated in detail by ENSREG group in the form of a recommendation of a detailed outline of the evaluation reports submitted by the operators and national reports processed and submitted by national nuclear safety regulators.

II.1 Activities of SÚJB after the accident at the Fukushima nuclear power plant

As with other countries operating nuclear power plants, the Fukushima Dai-ichi NPP accident triggered activities in the Czech Republic aimed at assessing the level of nuclear safety with regards to this accident.

SÚJB (State Office for Nuclear Safety) immediately appointed a group of experts who were given the task to evaluate the situation based on the analyses obtained from Japan and subsequently ensure communication and objective informing of the public about the situation while maintaining work-related communication with the representatives of the operator/licensee for the preparation of reports and information required by the European Commission in the time following the accident. With regards to the scope of work and the ensuring of independent analyses, SÚJB provided analytical expert support by Centrum výzkumu Řež s.r.o. (hereinafter referred to as Řež Research Centre) and an analytical group of selected employees of the National Radiation Protection Institute (NRPI). A website that allowed the public to ask questions about the events at the Japanese Fukushima NPP was launched in cooperation with Řež Research Centre. Both groups met twice a week in the first days following the accident. At the start, NRPI analytical group prepared a report twice a day on the current situation in Japan, reactions and stances in the world and on the current radiation status in our territory. As the situation in Japan became stabilized, the report frequency was reduced to two reports per week.

I.1.1 Monitoring and Radiation monitoring network

Following the instruction by SÚJB the frequency of sampling and analysis of volume activities in the atmosphere by the Central Laboratory of the Radiation Monitoring Network in NRPI increased as of 29th March 2011, while the monitored radio nuclides included: 131I, 134Cs, 137Cs, 132Te, 132I, 210Pb. All results of environmental factor monitoring were published at NRPI website (www.suro.cz) and also at SÚJB website (www.sujb.cz) with a reference. The central laboratory participated in monitoring food and animal feed imported from Japan (monitored for 131I, 134Cs a 137Cs). Based on the request of SÚJB, the Central laboratory provided free of charge measurements of internal contamination of persons returning from

Japan; travelers from other areas were charged a fee for the measurement (monitoring of 131I in the thyroid gland, 134Cs and 137Cs). Fees were also charged for measurements taken on non-food items sent from Japan (monitoring of 131I, 134Cs and 137Cs).

I.1.2 Communication on the part of SÚJB

In association with the development of events, SÚJB, or more specifically, NRPI communicated with the Government of the Czech Republic, the Czech embassy in Japan, the media, the public, relevant ministries and their subordinate organizations and, last but not least, with international organizations.

SÚJB was in daily contact with the Czech embassy in Japan from 18th March 2001 to 30th April 2011. It continuously assessed the radiation status and issued recommendations for embassy staff and Czech citizens in Japan. In March and April 2011 SÚJB made estimates of the irradiation of embassy employees as a result of the Fukushima accident and the final evaluation was sent to the embassy on 12th May 2011.

SÚJB organized an interdepartmental discussion forum concerning the events in Japan with the participation of the Office of the Government of the Czech Republic, the ministries and their subordinate organizations on 27th July 2011.

Since 14th March 2011 SÚJB has been receiving a large amount of querries regarding the situation in Japan. Information regarding the technological and safety characteristics of the damaged units and severe accidents were published on SÚJB website in cooperation with experts from Řež Research Centre and ÚJV Řež a.s. (Nuclear Research Institute – NRI) in the first days following the accident within the framework of communication with the public. Updated information concerning the development of the situation in Japan and in the Czech Republic was concurrently published every day together with the standpoints of SÚJB on individual fields (health effects of the doses of ionising radiation, iodine prophylaxis, current radiation situation in the territory of the Czech Republic, traveling to the third countries – namely Japan, inspections of food and non-food items from Japan, comments on breaking news, etc.). This interactive website platform proved very successful.

I.1.3 Communication with international organizations

In the exposed period following the accident SÚJB communicated bilaterally with international organizations IAEA, OECD and EC, including the ECURIE system.

Information on the implemented measures and their scope was reported to OECD and ECURIE to complete an overview of reactions of individual member countries. In the case of ECURIE, there was also an exchange of information concerning the current development of the situation.
The communication with IAEA consisted of the handover of information concerning the current situation development and international support for Japan.

Communication with EC (DG Health and Consumers - SANCO) regarding the results of measured food and animal feed coming from Japan was conducted via the State Agriculture and Food Inspection Authority. Communication with WHO regarding food monitoring in the following periods: 1st January 2010 to 11th March 2011 and 12th March to 14th September 2011 was conducted via the National Institute of Public Health.

SÚJB and NRPI participated in a domestic comment procedure to the implementing regulations of the European Commission (EC) on the stipulation of special conditions for the importing of animal feed and food in transit or shipped from Japan following the Fukushima accident.

I.2 Stress test implementation procedure

SÚJB transferred the EC request for the performance of stress tests to ČEZ a.s. as the operator of both nuclear power plants (Dukovany and Temelín) by means of a letter on May 25th 2011. ČEZ a.s. commenced the preparation of reports on the stress tests on 1st June 2011 by appointing a special group within the Production division consisting of employees from both power plants and by determining responsible guarantors for individual parts of both reports. The group included experts on nuclear safety, nuclear plant design and operation, accident response management, accident response planning, phenomenology of severe accidents, etc.

The implementation of stress tests itself took place in several stages:

- On 1st – 15th June information was collected from all relevant documents related to safety (safety analysis reports SARs, PSA studies, PSR documentation, rules for abnormal situations and accidents – EOP, SAMG and documents of the IAEA, WANO, etc.). At the same time on-site walk downs and checks of important systems and facilities were initiated at both power plants to verify their actual status. Based on the results of this inspection and analyses of safety documentation, the first versions of reports on stress tests were processed for both power plants and were evaluated for correctness and balance of information.

- On 15th – 22nd June a detailed evaluation and summary of the reports was prepared by expert departments of both NPPs and the reports were discussed in the Safety Committee of ČEZ a.s. The evaluation focused on the identification of weaknesses and on suggesting possible measures for the enhancement of resilience of both NPPs. Information exchange with other operators of VVER nuclear power plants also happened during this stage.

- On 15th July – 15th August – nuclear safety experts from ÚJV Řež a.s. commented on the reports with the focus on the field of evaluation of severe accidents.

- On 8th August 2011 progress reports on the stress tests of Dukovany NPP and Temelín NPP were approved by the Safety Committee of ČEZ a.s. and on August 15th 2011 both reports were made available to the State Office for Nuclear Safety (SÚJB) for evaluation.

- The SÚJB subsequently requested Centrum výzkumu Řež s.r.o. (Research Centre Řež), as a TSO, to provide evaluations of both reports and provide comments, which were then handed over to the authors of both reports within ČEZ a.s. to include them in the preparation of the final versions of the reports.

- On 15th September SÚJB sent a report on the progress of work on the stress tests to the EC.

- On 31st October ČEZ a.s. handed over the final reports on the results of stress tests of Dukovany NPP and Temelín NPP over to SÚJB.

In the course of processing the assessment, several work meetings took place with the responsible for stress tests personnel from other VVER-type NPPs within the framework of the "VVER club" (NPP Dukovany, Paks, Loviisa, Bohunice, Mochovce) and Kozloduj NPP and with partner supervisory bodies of countries operating NPP with VVER. Discussions with other VVER-type NPP operators outside the EU took place within the framework of WANO MC.

An independent assessment of results by the recognized external consultants in the field of nuclear safety, including ÚJV Řež a.s. and Westinghouse, was carried out with the aim of ensuring the objectivity of the assessment.

Evaluations carried out within the framework of stress tests, both on the operator and national regulator level, include:

- review of design requirements and their compliance,
- assessment of resilience, robustness against beyond design basis accidents (safety margins, diversity, redundancy, physical separation, etc.) and efficacy of the defence in depth system, including identification of cliff edge effects and possible measures for avoiding such cliff edge effects,
- identification of all means necessary to maintain the 3 fundamental safety functions (control of reactivity, fuel cooling, release prevention) and support functions (electricity power supply, transfer of heat to the ultimate heat sink) and consideration of effective possibilities for further enhancement of the defence in depth

The assessment covers all operational modes and states of the nuclear units. It particularly deals with impacts of events such as earthquake, floods, extreme weather conditions, loss of off-site power, station blackout and loss of the ultimate heat sink. A significant part of the report is the chapter called "severe accidents" which describes processes and strategies designed to cope with these events in various phases. The evaluation and conclusions of both final reports submitted by ČEZ a.s. go far beyond the scope of licence requirements stipulated by the legislation in force (Act no. 18/1997 Coll. and its executive regulations).

The results of evaluation of stress tests of Dukovany NPP and Temelín NPP are summed up in this National Report together with the summary statement of SÚJB.

I.3 Nuclear safety evaluation history in the Czech Republic

Nuclear safety evaluation for the purpose of building permit for nuclear facilities has a long tradition in the Czech Republic. In the 1970s the obligation to submit three types of safety reports within the land-use planning procedure, building permission procedure and final building approval procedure was included in the Building Act in former Czechoslovakia in relation to the construction of V-1 NPP in Jaslovské Bohunice. These included siting, preliminary and pre-operational safety analysis reports. The first generally binding legal regulations governing the evaluation of these reports by the state supervisory body, namely the Czechoslovak Atomic Energy Commission (ČSKAE), were issued in the 1970s, and included:

- ČSKAE directive no. 2 on Nuclear Safety Assurance within the Process of Designing Nuclear Installations (1978)
- ČSKAE directive no. 4 on Nuclear Safety Assurance within the Siting of Nuclear Installations (1979)

At that time, the elementary criterion base was based on the practical experience of countries with a developed nuclear power industry and on the recommendations of the International Atomic Energy Agency based in Vienna.

The nuclear legislation of the Czech Republic was significantly amended by the issue of Act no. 18 on the Peaceful Utilisation of Nuclear Energy in 1997 (the Atomic Act) and its implementing regulations, namely:

- SÚJB decree no. 132/2008 Coll., on Quality System in Activities Related to the Utilization of Nuclear Energy and in Radiation Activities and on Quality Assurance in Classified Equipment with Respect to its Categorisation into Safety Classes
- SÚJB decree no. 215/1997 Coll., on Criteria for Siting Nuclear Facilities and Very Significant Ionising Radiation Sources
- SÚJB decree no. 106/1998 Coll., on Nuclear Safety and Radiation Protection Assurance during Commissioning and Operation of Nuclear Facilities,
- SÚJB decree no. 195/1999 Coll., on Basic Design Criteria for Nuclear Installations with Respect to Nuclear Safety Radiation Protection and Emergency Preparedness**,**
- 309/2005 Coll., on Assurance of Technical Safety of Classified Equipment,
- SÚJB decree no. 185/2003 Coll., On the Decommissioning of Nuclear Installations or Category III or IV Workplaces,
- SÚJB decree no. 146/1997 Coll., as amended by SÚJB decree no. 315/2002 Coll., Specifying Activities Directly Affecting Nuclear Safety and Activities Especially Important from the Radiation Protection Point of View, Requirements Regarding Qualification and Professional Training, Method to be used for Verification of Special Professional Competency and for Issuing Authorisations to Selected Personnel and the Form of Documentation to be Approved for the Licensing of Expert Training of Selected Personnel,
- SÚJB decree no. 307/2002 Coll., on Radiation Protection, as amended by SÚJB decree no. 499/2005 Coll.
- SÚJB decree no. 318/2002 Coll., on the Details of Emergency Preparedness of Nuclear Facilities and Workplaces with Ionising Radiation Sources and on Requirements Regarding the Content of the On-site Emergency Plan and Emergency Rules, as amended by Decree no. 2/2004 Coll.,
- SÚJB decree no. 319/2002 Coll., on Performance and Management of the National Radiation Monitoring Network, as amended by SÚJB decree no. 27/2006 Coll.
- SÚJB decree no. 193/2005 Coll., on the Setting-up of a List of Theoretical and Practical Areas Which Constitute the Content of Education and Preparation Required in the Czech Republic for the Performance of Regulated Activities Belonging to the Competence of the State Office for Nuclear Safety (SÚJB),
- Government order no. 11/1999 Coll., on Emergency Planning Zone.

This legislative basis is supplemented with a range of safety guides which are not legally binding havingcharacter of recommendations and which SÚJB has been issuing on the basis of the harmonization of legislation pursuant to the reference levels set by the Western European Nuclear Regulators' **Association** (WENRA). All WENRA reference levels were thus included in the legislation. Recommendations included in the guides will be implemented in the legally binding legislation, i.e. into acts and decrees, in the near future. The stress test issues are mainly dealt with in the following instructions:

- **On the Requirements Concerning Nuclear Facility Designs, BN – JB – 1.0**
- **Probabilistic Safety Assessment, BN – JB – 1.6**
- **Selection and Assessment of Design Basis and Beyond Design Basis Events and Risks, BN – JB – 1.3**
- **Requirements for the Implementation of OEP and SAMB Type Operating Regulations, BN – JB – 1.11**

Requirements included in these guides are used as conditions for decisions which SÚJB issues to the licensees within the framework of its supervisory activities.

I.4 Periodic safety review (PSR)

Besides the permit procedure in connection with the construction of new nuclear installations there is also a long-term practice of periodic safety review (PSR) of NPPs at 10-year intervals implemented in the Czech Republic. To implement them, the instruction “Periodic safety review BN – JB – 1.2.” was issued by SÚJB. The periodic safety review evaluates to what extent the systems, structures and components of a nuclear installation (both individually and as a whole including their operators) comply with the current safety requirements included in the legal regulations of the Czech Republic, recommendations issued by WENRA and IAEA and international practice, and to what extent the original design bases which served as the basis for decisions of SÚJB in approval of the siting, construction and operation of a nuclear installation remain in effect. The result of a PSR is a set of measures for maintaining, or eventually enhancing, safety with the aim of ensuring the correspondent level of safety of a nuclear installation for the whole period of its operation until the next periodic safety review or eventually until the end of its usable lifetime.

The last periodic nuclear safety review of Dukovany NPP was carried out in 2006 and 2007, after 20 years of operation. This was an in-depth inspection of conformity with the requirements of both national and international legislative documents, requirements of WENRA defined in a document called “Reactor Safety Reference Levels” and other international recommendations in IAEA documents (Safety Guides). A comprehensive assessment within the framework of the PSR identified similar safety enhancement opportunities to those stated in this report. Some of them (increase of resilience of NPP Dukovany against the consequences of severe accidents, part of which is the increase of hydrogen removal system capacity by installing hydrogen recombiners for severe accidents and preparation for the flooding of the reactor shaft) are already being prepared for implementation and they would be implemented even without this new evaluation. The PSR expects the implementation of the approved measures by 2015, i.e. by the time of the renewal of permission for future operation of Dukovany NPP.

A periodic safety review of Temelín NPP was carried out in 2008 and 2009. Its technical content and scope was identical to the one performed in Dukovany NPP. Its conclusion also included a range of measures to increase the resilience of the Temelín NPP design against the consequences of severe accidents. These measures are currently being implemented, e.g. increase of hydrogen removal system capacity. The PSR of Temelín NPP expects implementation of a majority of the approved measures by 2015, in some justified cases even by the next PSR (in 2018).

I.5 International VVER evaluation within the framework of the IAEA Extrabudgetary Programme

Both NP designs underwent an extraordinary international evaluation process within the framework of IAEA Extrabudgetary Programme focusing on VVERs which took place in 1991 – 1997.

The output of this extensive programme included the Final Project Report (1999) and ‘safety books’ for individual types of VVERs. The National Report of the Czech Republic for the Convention on Nuclear Safety (2010) documents solutions of safety issues mentioned in the report for Dukovany NPP and Temelín NPP as follows:

- Dukovany NPP – 65 safety issues have been solved, 9 safety issues are being dealt with
- Temelín NPP – 70 safety issues have been solved, 1 safety issue is being dealt with

A lot of these "safety issues", namely Category III (nuclear safety important) directly concern the stress test issue. The following can be mentioned as an example:

Category III "Safety issues" for NPPs with VVER-440/213 (state at Dukovany NPP):

G 02 Qualification of components - being implemented
CI 02 Non-destructive testing - solved
S 05 ECCS sump screen blocking - solved
S 13 SG feedwater system vulnerability - solved
C 01 Increase of bubbler condenser system strength under LOCA conditions - solved
IH 02 Fire prevention - solved
IH 07 Internal hazards due to high energy piping rupture – being implemented
EH 01 Increase of seismic robustness – being implemented

Category III "Safety issues" for NPPs with VVER - 1000 (state at Temelín NPP):

G 02 Qualification of components – being implemented
RC 02 Control rod insertion reliability / Fuel assembly deformation - solved
CI 01 RPV embrittlement and its monitoring - solved
CI 02 Non-destructive testing - solved
CI 04 Steam generator collector integrity - solved
CI 06 Steam and feedwater piping integrity - solved
S 05 ECCS sump screen blocking - solved
S 09 Qualification of steam generator relief and safety valves for water flow - solved
I&C 08 Reactor vessel head leak monitoring - solved
EI 05 Emergency battery discharge time - solved
IH 02 Fire prevention - solved

The solved "safety issues" include a range of components and systems which are related to prevention or to the focus of this report, i.e. mitigation of severe accidents, e.g.: hydrogen removal (S15), diesel generators reliability (EL02), accumulator batteries (EL 03), bubbler condenser system behavior (C01, C02, C03), seismicity (EH01, EH02), severe accidents - SAMG (AA 09), blackout (AA 14) and loss of ultimate heat sink (AA 15).

I.6 International missions at Dukovany NPP and Temelín NPP

In order to ensure an objective evaluation of the actual state of nuclear safety of Dukovany NPP and Temelín NPP, international missions were invited to both sites, namely IAEA missions, which evaluate the selected fields in terms of their safety requirements and guides and, at the same time, in terms of best international practice. This included the following missions:

Dukovany NPP:
- 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)
- Safety Issues of VVER 440 Resolution Review, IAEA (1995)

- Nuclear Insurance Pool, Marsh & McLennan (1996)
- Nuclear Insurance Pool, organized by Gradmann & Holler (1996)
- Nuclear Insurance Pool, ČJP (1997): inspection of insurance risks (an insurance contract on the coverage of liability risks was signed between ČEZ, a. s., and ČJP in December 1997),
- Peer Review, WANO (1997)
- International Peer Review Service (IPERS), IAEA (1998)
- International Physical Protection Advisory Service (IPPAS), IAEA (1998)
- Nuclear Insurance Pool, ČJP (2000)
- EMS certification audit, Det Norstke Veritas (2001): final assessment pursuant to the requirements of ISO 14001 standard
- Operational Safety Review (OSART), IAEA (2001)
- Operational Safety Review Follow-up, (OSART FU), IAEA (2003)
- Nuclear Insurance Pool, ČJP (2006)
- Peer Review, WANO (2007)
- Safety Assessment Long Term Operation Review (SALTO), IAEA (2008)
- Peer Review Follow-up, WANO (2009)
- Operational Safety Review (OSART), IAEA (2011).

Temelín NPP:
- Site Safety Review, Design Review, NUS Halliburton (1990): evaluation of the site, safety systems, active zone design and safety analyses
- Pre–Operational Safety Review (Pre-OSART), IAEA (1990)
- Pre–Operational Safety Review Follow up, IAEA (1992)
- Quality Assurance Review IAEA (QARAT) (1993)
- Leak Before Break Application Review, IAEA (1993 – 1995)
- Fire Safety IAEA (1996)
- International Peer Review Service – PSA 1, PSA 2, IAEA (IPERS), (1995 – 1996)
- Safety issues of VVER 1000 Resolution Review, IAEA (1996)
- Physical Protection Assurance – IPPAS, IAEA (1998)
- Operational Preparedness and Plant Commissioning Review, IAEA (2000)
- Operational Safety Review (OSART) IAEA (2001)
- Safety Issues of VVER 1000 Resolution Review Follow up, IAEA (2001)
- International Physical Protection Advisory Service (IPPAS), IAEA (2002)
- Site Seismic Hazard Assessment – expert mission, IAEA (2003)
- International Probabilistic Safety Assessment Review, IAEA (IPSART) (2003)
- Operational Safety Review Follow-up (OSART FU), IAEA (2003)
- Peer Review, WANO (2004)
- Peer Review Follow-up, WANO (2006)
- WANO Peer Review (2011).

All the above mentioned missions concluded, regardless of the range of formulated recommendations and suggestions, with positive statements confirming the adequacy of the nuclear safety level of both NPPs and examples of above-standard safety measures (good practices).

I.7 Basic legislative procedures and requirements

The basic legislative procedures and requirements to ensure nuclear safety in the Czech Republic are governed by Act No. 18/1997 Coll., on the Peaceful Use of Nuclear Energy and Ionizing Radiation (the Atomic Energy Act), and implementing regulations. Section 9 of Act No. 18/1997 Coll. specifies the main phases of the licensing process for nuclear installations:

a) Location of the nuclear installation and spent fuel storage
b) Construction of the nuclear installation
c) Operation of the nuclear installation
d) Renovations and changes that influence nuclear safety, radiation safety, physical safety and emergency readiness of the nuclear installation
e) Individual phases of removing nuclear installations from operation.

The applicant requesting permission for the above activities must attach the corresponding safety documentation to the request.

A permit for a nuclear facility or nuclear waste warehouse requires an Initial Safety Report that contains:

1. The characteristics and proof of suitability of the selected location based on the criteria for the locations of nuclear installations and nuclear waste warehouses, as defined by the implementing regulation.
2. The characteristics and preliminary evaluation of the design's concept in terms of the requirements defined by the implementing regulation for nuclear safety, radiation safety and emergency readiness.
3. A preliminary evaluation of the impact the operation of the proposed facility will have on the employees, population, and the environment.
4. A proposal of a concept for the safe termination of operation.
5. An evaluation of the quality guarantees for the selection of a location, method of ensuring quality in the construction process, and principles of quality assurance in the subsequent stages.

Before a building permit can be issued, it is necessary to submit a Preliminary Safety Report that contains:

1. Proof that the solution proposed in the design complies with the requirements on nuclear safety, radiation protection, and emergency preparedness, as defined by the implementing regulation.
2. A safety analysis and analysis of the possibility of unauthorized access to nuclear material and sources of ionizing radiation, as well as the evaluation of the potential consequences for the employees, population, and the environment.
3. Data regarding the expected lifetime of the nuclear facility or very significant source of ionizing radiation.
4. An evaluation of the production of radioactive waste and handling of this waste during activation and operation of the facility or workplace in question.
5. A concept of the safe termination of operation of the facility or workplace in question, including the liquidation of radioactive waste.
6. A concept for dealing with spent nuclear fuel.

An approval of the SÚJB for operation is issued after evaluating a Pre-Operation Safety Report that contains:

1. Addenda to the Pre-Operation Safety Report and other additional documentation requested before the permit for the first delivery of nuclear fuel to the reactor can be issued, as well as information related to the changes implemented after the first delivery of nuclear fuel.
2. An evaluation of the results of the previous activation stages (putting into operation).
3. Proof that the previous decisions and conditions of the SÚJB have been fulfilled.
4. Proof of the readiness of the facility and personnel for operation.
5. A schedule of operation.
6. Updated limits and conditions for standard operation.

The term **"nuclear safety"** is specified in the definition of this term that can be found in the Atomic Energy Act, which defines nuclear safety for the purposes of this Act as the *"state and capacity of a nuclear facility and persons operating the nuclear facility to prevent uncontrolled development of a fission chain reaction or prohibited release of radioactive substances or ionizing radiation into the environment **and limit the consequences of accidents**"*.

Specific requirements on the designs of nuclear facilities can be found in Regulation No. **195/1999** Coll. of the SÚJB, on the Requirements on Nuclear Installations for the Assurance of Nuclear Safety, Radiation Protection and Emergency Preparedness. With respect to the external risks, the regulation defines the following requirements in terms of the protection against events caused by natural conditions or human activity outside of the nuclear installation:

Section 10 - Protection against Events Caused by Natural Conditions or by Human Activity Outside of the Nuclear Installation

(1) The components important for the nuclear safety of a nuclear installation shall be designed in such a way that, under conditions of such natural events that could reasonably be considered (earthquakes, windstorms, floods, etc.) or under events resulting from human activities outside of the nuclear installation (aircraft crash, explosions in the vicinity of the plant etc.), it is possible
- *a) to shut down the reactor safely and keep it in a sub-critical state,*
- *b) to transfer the residual output of the reactor for a sufficiently long period,*
- *c) to assure that any radioactive leakage does not exceed the limit values stipulated by the special Regulation*

(2) When designing a nuclear installation, the following factors need to be considered:
- *a) the most severe natural phenomena or events that have been historically reported for the site and its surroundings, extrapolated with a sufficient margin for the limited accuracy (uncertainties) in values and in time,*
- *b) a combination of the natural phenomena or effects resulting from the human activities and the consequent emergency conditions.*

Regulation No. **195/1999** Coll. of the SÚJB contains specific technical requirements for the reactor's cooling systems, protective containment, power supply systems and their backup systems, including the requirements for them to function under ordinary and extraordinary operation, as well as in emergencies, which also includes external events that can reasonably be considered on the basis of historical records.

The key provisions containing the technical specifications of stress tests are: Provision on the transfer of heat and provision on the power supply backup systems. The transfer of heat is, among other things, addressed in the following provision:

Section 25 – Residual Heat Removal System

(3) The residual heat removal system shall ensure that, during a reactor shutdown, the design limits of the fuel elements and of the primary circuit are not exceeded.

(4) The key components of the residual heat removal system must have sufficient backup, suitable connections, the ability to disconnect parts of the system, a system for detecting leaks, and the capacity to capture leaks so that the system operates reliably even in the event of a single failure.

Section 26 – Emergency Core Cooling System

The emergency core cooling system shall ensure:

a) *reliable cooling of the core under emergency conditions caused by a loss of coolant, so that*
   1. *the temperature of the fuel cladding does not exceed the values stipulated by the design limits,*
   2. *the energy contribution of the chemical reactions (cladding, water, hydrogen release) does not exceed the acceptable value,*
   3. *there are no changes to the fuel elements, fuel assemblies or reactor internal components that could influence the efficiency of the cooling,*
   4. *the residual heat is being removed for a sufficiently long period,*

b) *its sufficient redundancy, suitable interconnection, ability to disconnect parts of system, a system for detecting leaks. and the capacity to capture leaks, so that the system operates reliably even in the event of a single failure.*

The requirements on power supply are specified in the following provisions:

Section 29 – Power Supply Systems

(1) *The discharge of the power output from a nuclear installation and the system for internal (own) power supply shall ensure that*
   (a) *external and internal failures on the power mains have a minimal impact on the operation of the reactor and heat removal systems,*
   (b) *systems in the power plant important for the operation can be supplied from two different sources (on-site generator and electrical grid).*

(2) *The power supply system for the control and protective systems of the primary circuit, the residual heat removal systems, the emergency cooling system and for the Containment System must be compatible with the emergency power supply, i.e. these systems must be redundant even when their generators or power mains are not in operation. The operation and protection systems must have an uninterrupted power supply.*

Section 30 – Redundancy of the Power Supply Systems

(1) Systems that must be redundant to ensure nuclear safety shall be supplied from power sources in such a way that they are functionally independent, i.e. the power supply systems and their sources must be independent. If the number of sources is lower than the number of independent systems, the design must demonstrate that this does not reduce their reliability.

(2) If a single failure of the supplied systems cannot affect their function, a single failure of the electrical system or source can be tolerated.

(3) If nuclear safety requires the operability of a certain system, the power source supplying this system must provide the necessary power supply even in the event of a simple failure.

Section 31 Emergency Power Sources

(1) *The systems that must be connected to a power supply without an interruption (Category I appliances) will be connected to power sources that supply power immediately (batteries with alternators).*

(2) *The sources and supply systems that are launched in case of an emergency with a delay (Category II appliances) must be running and providing the necessary output in a time period shorter than the time period necessary for launching Category II appliances.*

(3) *The capability to carry out functional testing of the emergency power sources shall be ensured.*

The containment, as the last barrier, plays the key role in preventing radioactive leaks in case of severe accidents. For the containment system (among other things), Regulation No. 195/1999 Coll. of the SÚJB contains the following provisions (in this case, emergency conditions are conditions for a design accident):

Section 33 Design principles

(1) The Containment System consists of a hermetic envelope dimensioned for all design basis accidents, closing elements, pressure and temperature reduction systems, and venting and filtration systems.
(2) The Containment System must maintain the required level of tightness in case of an emergency, and for a sufficient length of time after the emergency.
(3) The Containment System must deliver the required functionality for the maximum values of pressure, underpressure and temperature possible during design basis accidents. It is necessary to consider the influence of the pressure and temperature reduction systems inside the containment, as well as the influence of other potential sources of energy, penetrations and access openings, uncertainties of the simulation models, results of experiments and experience from the operation.
(4) The Containment System must comply with the requirements specified for protection against external effects according to Section 10.
(5) The components of the Containment System must ensure functioning of the Containment System and have only a limited impact on other systems and components important for nuclear safety.

Section 35 Hermetic Envelope Pressure Test
Prior to putting the nuclear installation into operation the hermetic envelope must be subjected to a pressure test to prove its integrity under test pressure, which is higher than the design pressure.
Section 41 Hermetic Envelope Pressure Reduction and Heat Removal System
(1) The hermetic space must be combined with a pressure reduction and heat removal system, capable (together with other systems) of quickly reducing the pressure and temperature in the hermetic space after an emergency event that includes the release of matter and energy. This system must also ensure that the permissible values are not exceeded.
(2) The system must be reliable, with sufficient backup and functional diversity to guarantee that it remains functional in the event of a single failure.

Section 42 Other Systems of the Protective Envelope

(1) The Containment System must include systems controlling fission products and substances that could be released from the Containment System in case of an accident. These systems must be able (in combination with other systems)
  (a) to reduce the volume activity and modify the composition of fission products
  (b) to control the volume concentration of explosive substances to guarantee the integrity of the hermetic envelope and reduce the amount of escaping radionuclides.
(2) The key components of these systems must be redundant, allowing them to operate even in the event of a single failure.

I.8 Emergency preparedness

National legislation defining the requirements on emergency preparedness in accordance with the IEAE, in particular:
- Safety Standards GS-R-2: Preparedness and Response for Nuclear and Radiation Emergencies, 2002
- Safety Standards GS-G-2.1: Arrangements for Preparedness for Nuclear and Radiation Emergencies, 2007.

The provisions of Section 2 of the Atomic Energy Act define the basic terms – emergency preparedness, radiation accident, extraordinary radiation event, accident radiation, zone of emergency planning and emergency plan.

Based on Section 3 of the Atomic Energy Act, the SÚJB, within the scope of its authority:

- approves internal emergency plans and changes to these plans after disputing links to external emergency plans; the approval of an internal emergency plan is a necessary condition for issuing the permit to commence the operation of and operate a nuclear installation,
- defines the zone of emergency planning based on a request from the holder of the permits,
- manages the activities of the national radiation monitoring network and controls its functions from the headquarters,
- ensures the functioning of the emergency coordination centre and international exchange of data about radiation situations,
- using national radiation monitoring network, and based on the evaluation of the radiation situation, creates an information basis for making decisions about measures leading to reducing or preventing radiation events in case of a radiation accident,
- is obliged to release adequate information to the general public about its activities, unless such information is confidential, and annually, execute a report about its activities to be submitted to the government and the public.

In Section 4, the Atomic Energy Act specifies the principles for carrying out radiation activities and limiting accident radiation. The principles of preventing or reducing radiation in case of a radiation event and the exposure of persons to radiation who take part in response activities, are elaborated in detail in Implementation Regulation No. 307/2002 Coll. of the SÚJB, on Radiation Protection.

In Section 17, the Atomic Energy Act assigns the general duty to the holder of the permits to ensure emergency preparedness and verify it to the extent required by the permits, and to inform the SÚJB of every change relevant from the point of view of emergency preparedness, including all changes crucial for the issuance of the permit.

The provisions of Section 18 of the Atomic Energy Act also define the following duties, among others, for the holders of the permit:

- monitor, measure, evaluate, verify and record the values, parameters and observations relevant for emergency preparedness in the scope specified by the implementing regulations,
- keep and maintain a register of ionizing radiation sources, objects, material, activities, values and parameters and other data relevant for emergency preparedness; the registered data must be submitted to the SÚJB in the form specified in the implementing regulation,
- ensure nonstop compliance with the requirements of emergency preparedness, including methods of verification.

The provisions of Section 19 of the Atomic Energy Act define the following duties of the holder of the permit in the event of a radiation accident within the scope and employing the methods specified by the internal emergency plan approved by SÚJB:

- inform the state administration authorities, SÚJB and other relevant bodies specified in the internal emergency plan immediately of any radiation event or suspected radiation accident,
- in the event of a radiation accident, warn the population in the emergency planning zone,
- immediately remove the consequences of a radiation accident from the premises and take measures to protect the employees and other persons from ionizing radiation,
- monitor the radiation dosage received by the employees and other persons and the amount of radionuclides and ionizing radiation released into the environment,
- inform the authorities of the outcomes of the monitoring, of the actual and expected development of the situation, of the measures taken to protect the employees and the

population, of the measures taken to remove the consequences of a radiation accident and of the actual and expected radiation dosage received by involved perSÚJB

- monitor and regulate the radiation exposure of employees and persons participating in the liquidation of a radiation accident in the premises,
- cooperate in the effort to remove the consequences of a radiation accident in the installment,
- in the event of a radiation accident, participate in activities related to the operation of the national radiation monitoring network

Details and requirements in the area of emergency preparedness in case of extraordinary events (radiation accidents and emergencies) are also specified in the regulations implementing the Atomic Energy Act:

- ***Regulation No. 318/2002 Coll. of the SÚJB***, as amended by Regulation No. 2/2004 Coll. of the SÚJB, contains detailed specifications ensuring the emergency preparedness of nuclear installations with sources of ionizing radiation and requirements on the contents of the internal emergency plan and emergency rules.
  In accordance with this regulation, the operator of a nuclear power plant (holder of the permit) is obliged to create organizational and personnel precautions that would allow the personnel of the nuclear power plant to react immediately in case of an extraordinary event and carry out the previously planned activities designed to suppress the negative effects of this situation.
- ***Regulation No. 307/2002 Coll. of the SÚJB, on Radiation Protection, as amended by Regulation No. 499/2005 Coll. of the SÚJB***
- ***Regulation No. 319/2002 Coll. of the SÚJB, on the Functions and Organizations of the National Radiation Monitoring Network, as amended by Regulation No. 27/2006 Coll. of the SÚJB.***

I.9 Requirements for beyond basis design accidents

For a relatively long period of time, the SÚJB has had an amendment to Regulation No. 195/1999 Coll. in a nearly finished phase. This regulation defines the safety objectives and safety principles, as well as the requirements for nuclear installations and reactors with an output exceeding 50 MWt. This amendment is currently published as the Manual of the SÚJB – on the Requirements for Nuclear Installation Design BN-JB-1.0. This amendment includes, in accordance with safety manual IAEA NS-R-1, events of the “extended design conditions” type, and declares specific requirements for beyond design basis accidents (BDBA).

In connection with the beyond design basis accidents the manual contains, among other provisions, the following:

Safety evaluation

(36) Consideration shall be given to severe beyond design basis events (extended design conditions), using a combination of engineering judgment and probabilistic methods, to determine those events for which reasonably practicable preventive or mitigatory technical and organizational measures can be identified.

(37) An analysis of these beyond design basis accidents need not involve the application of conservative engineering practices used in setting and evaluating design basis accidents, but rather should be based upon realistic or best-estimate assumptions, methods and analytical criteria (it is not necessary to apply the single-failure criterion; in addition, the involvement of systems not classified as safety systems can be considered, etc.).

(38) Those developments and radiation consequences of severe accidents must be considered that cannot be practically excluded:

- *to identify practicable measures preventing the occurrence and development of accidents and to manage mitigation of the consequences,*
- *as a basis for manuals designed for coping with accidents and for staff training ,*
- *as a basis for executing plans for protecting the staff and the population, and implementing mitigating measures to reduce the impact of radioactive leaks on the staff, population and the environment,*

Reactor pressure and cooling circuit

(82) The proposal of the primary circuit design must ensure, for the staff, technical means and the ability to implement organizational measures to prevent reactor meltdown under high pressure in the core cooling circuit under emergency conditions of heavy accidents.

Protective Containment System

(108) The design criteria (including limits for the temperature and pressures inside the protective Containment System and its hermicity) must be specified to protect and ensure the functions of the Containment System, and the design must guarantee that these criteria are complied with:

- *in case of design basis accidents for a period of time sufficient for reaching a safe and stabilized state,*
- *in case of a severe accident, at least for a period sufficient for the realization of measures according to the specific legal provisions).*

(117) The design must guarantee that the loss of the safety functions of the Containment System is virtually impossible. Procedures, technical means and organizational measures for maximum protection of the integrity and functionality in beyond design basis accidents, including severe accidents, must be available. These procedures, means, and measures will minimize the consequences of possible overpressure, overheating, damage from exploding gases or melt from degraded debris from the core, the release of radioactive substances in liquid form, such as vapor, melt from the core, etc.

I.10 Requirements on accident management

Severe accident management guidance (SAMG) was first implemented in the Czech Republic during the activation of the Temelín NPP, using the experience of Westinghouse. The requirements of the SÚJB on accident management are currently summarized in the SÚJB Manual – Requirements on the Implementation of EOP and SAMG-type Operating Instructions, BN – JB – 1.11. This manual specifies the requirements on a program for accident management, including operating instructions to cope with design basis and beyond design basis accidents, including severe accidents. The manual includes requirements on the format, scope and content of such instructions, including maintenance and staff training. Most of the requirements from this manual are based on IAEA Safety Standard – Severe Accident Management Programme NS-G-2.15. The obligation to implement EOP- and SAMG-type operating instructions is based on the following requirements in the manual:

(3.17) Staff carrying out measures required by accident management must have suitable operating instructions in the form of a regulation or manual.

(3.45) The program for accident management must be carried out in the following steps:

a) identification of any vulnerability (weakness) of the NPP in case of accidents to find mechanisms through which critical safety functions and barriers preventing the release of fission products may be challenged.
b) the plant's capabilities under challenges to critical safety functions and fission product barriers should be identified, including capabilities to mitigate such challenges, in terms of both equipment and personnel.
c) suitable accident management strategies and measures should be developed, including hardware features, to cope with the vulnerabilities identified;
d) procedures and guidelines to execute the strategies should be developed.

(3.25) Manuals for severe accident management must consider specific threats related to reactor shutdown modes and long-term shutdowns, involving open containment. The manuals must include potential damage to the nuclear fuel in the reactor vessel, as well as in the spent fuel storage pools. Because general maintenance is carried out during planned shutdown modes, the modes must be focused primarily on staff safety.

(3.32) Implemented EOP and SAMG will be an integral part of accident measures in the NPP. According to the SAMG, the emergency response organization (ERO) is responsible for carrying out activities according to the SAMG. The functions and responsibilities of the members of the ERO involved in accident management must be clearly defined and coordinated.

In the general practice of the SÚJB, the above requirements are transformed into binding conditions of decisions issued by the SÚJB regarding a permit to operate. For example, the permit to operate the units of the Dukovany NPP issued in 2005 and 2007 contained the following condition:

"The Applicant will continue to develop an accident management program, including beyond design basis accidents, and the results of these efforts are to be reported to the SÚJB by the end of the 1st quarter of the following year."

Similarly, permits from 2004 and 2005 to operate the 1st and 2nd units in the Temelín NPP contain the following condition:

"The Applicant will update the Severe Accident Management Guide (SAMG), including instructions for the unit's Control Room and Technical Support Centre. The SÚJB is to be informed of all updates once annually, by the end of the 1st quarter of the following year."

These conditions have been fulfilled by both power plants.

II NPP Dukovany

II.1 General data about the site/plant

II.1.1 Brief description of the site's characteristics

The nuclear power plant Dukovany (NPP Dukovany) is situated southwest of the city of Brno, on a leveled area with elevation of the ground at 389.3 meters above sea level that is bordered by a deeply eroded valley of the river Jihlava on the north. The surrounding elevations of the ground vary from 370 to 395 meters above sea level. Mohelno, Dukovany, Rouchovany and Slavětice are the nearest municipalities situated at distances from 3 to 5 km. A class 2 road, no. 152 Moravské Budějovice – Brno, runs northeast of the JE area. NPP Dukovany is connected to the railway network from the east by means of a railway siding from the connecting station Rakšice on the Czech railways line Moravský Krumlov - Brno.

Four nuclear units (utilizing some shared equipment) are operated in the site of NPP Dukovany. The units are identical, joined constructionally into double units.

A pumped-storage power plant Dalešice is located on the river Jihlava; this power plant is also used as a reservoir of water for the nuclear power plant. Electrical energy is fed from the nuclear power plant to a 400kV switching station Slavětice.

The site of NPP Dukovany also includes two spent nuclear fuel storage packet sets (ISFS/SFS). Spent nuclear fuel is stored in containers CASTOR cooled by natural circulation of air in the ISFS/SFS. Due to the passive principle of cooling there is no risk of loss of the safety function of these containers after occurrence of an initiating accident and ISFS/SFS is therefore not considered in this safety and safety margins assessment.

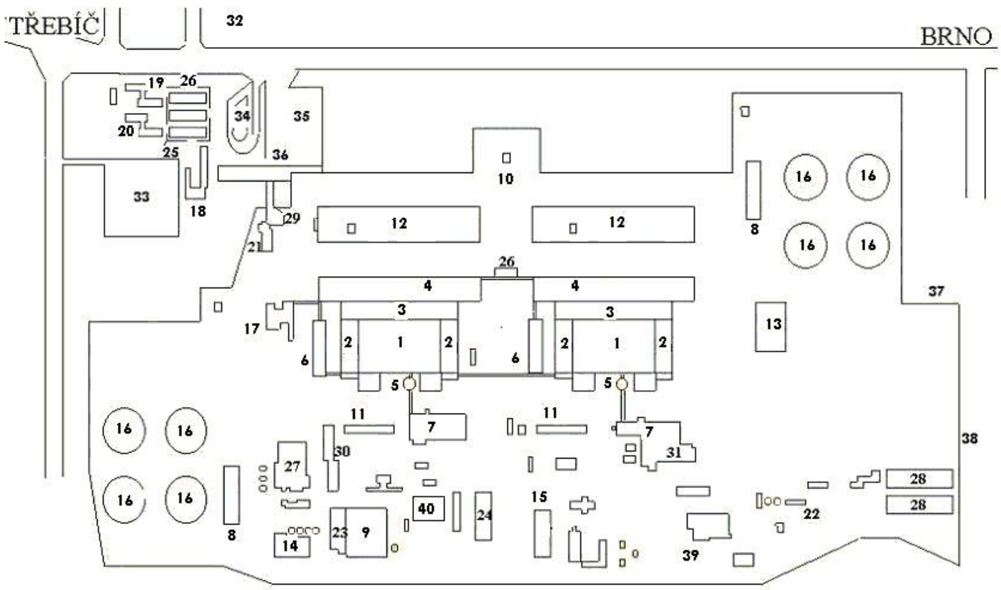


Fig. 1: Arrangement of significant objects of NPP Dukovany

1 – Reactors building, 2 – Transversal auxiliary floor, 3 – Longitudinal auxiliary floor, 4 – Machine room, 5 – Ventilation stack, 7 – Building of auxiliary active plants, 8 – Central pumping station, 10 – Industrial water treatment plant, 11 – Diesel sets station, 12 – 400kV switching station, 13 – Intermediate spent nuclear fuel storage packet set and spent fuel storage packet set 14 – Chemical water treatment plant, 15 – Compressor stations and cold source station, 16 – Cooling tower, 17 – Administrative building 1 (ECC cover), 28 – Radioactive waste repository, 29 – Porter's lodge, 30 – Plant fire brigade, 32 – Waste water treatment plant and two holding tanks

ČEZ a.s., Duhová 2/1444, 140 53 Praha 4 is the permit holder for the operation of all the nuclear equipment in the site. The currently valid permit for operation of NPP Dukovany was issued for the first unit as Decision of SÚJB, reg. no. 24273/2005 of December 16, 2005, for the second unit as Decision of SÚJB, reg. no. 55714/2006 of December 8, 2006, for the third unit as Decision of SÚJB, reg. no. 30852/2007 of December 10, 2007 and for the fourth unit as Decision of SÚJB, reg. no. 30853/2007 of December 10, 2007. All permits are valid for 10 years.

II.1.1.1 Main characteristics of the units

Individual reactor units of NPP Dukovany include pressurized-water reactors VVER 440 (type V-213č) with power 1,375 MWt, resp. 1,444 MWt. The units were put into operation in the years 1985 to 1987.

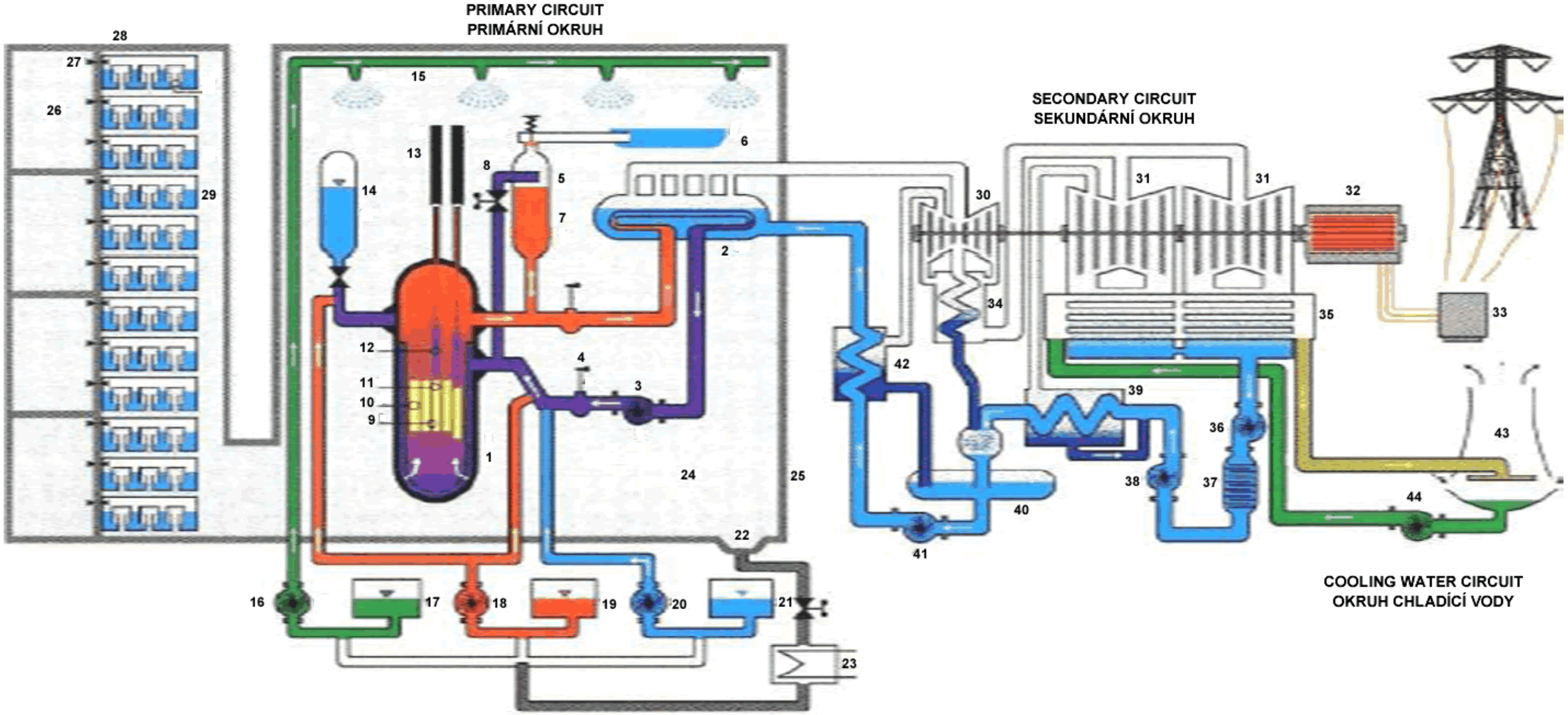
The reactor (resp. the reactor core) is cooled and moderated using primary circuit water that is pumped through the core using circulation pumps. Heat absorbed in the coolant is delivered to secondary circuit water in steam generators after passing through the reactor. Primary circuit pressure is maintained by a pressurizer. The reactor cooling system (primary circuit) consists of six loops of circulation piping with a main circulation pump (MCP) and horizontal steam generator (SG) with optional separation of leaking MCP or SG using main isolation valves and also by a volume compensation system.

The reactor and main primary circuit components are located in a robust hermetic zone – containment – that is formed by a reinforced concrete structure with hermetic liner and that forms a barrier against release of radioactive substances into the surroundings. The containment is located inside the reactor building that continues above the floor at the level of 18.9 m as a steel roof structure. Pools for storage of spent fuel (SFSP) are located in the reactor building; fuel from the reactor core is removed into these pools and after decrease of its residual power it is then transported continuously to ISFS/SFS in containers CASTOR.

The secondary circuit consists of two turbine generators for one unit with condensation, regeneration, feed water and a steam piping system. The secondary circuit is connected to circulating cooling water systems and technical water systems with four cooling towers for the main production unit (MPU) that are sometimes called a tertiary system.

Residual heat is removed during operation into the atmosphere through steam generators, main condensers, and during shutdown through steam generators, technological condensers, essential service water system (ESW) and cooling tower. Heat removal into the atmosphere is provided by an independent system of water piping with natural draught in cooling towers. The pumping stations for essential service water are designed as a separate construction for the double unit; there are therefore two pumping stations of ESW in the power plant area.

Redundancy of active safety systems is 3 x 100%; the systems are mutually independent and separated physically. Redundancy of passive safety systems (hydroacumulators inside the containment) is 2 x 100%. Seismic resistance of all redundant safety systems including power supply and control systems and other auxiliary systems is ensured. Backup sources for power supply systems and control systems are mutually independent, separated physically and seismic-resistant (subject to qualification as for safety systems). The design includes diversified systems to ensure fulfillment of three basic safety functions: 1) ensuring of reactor shutdown (subcriticality), 2) heat removal (cooling down) and 3) confinement - elimination of leakage (barriers and isolation of the containment).



1 – Reactor, 2 – Steamgenerator, 3 – Main coolantpump, 4 – Main isolating valve, 5 – Pressurizer, 6 – Bubble condenser, 7 – Pressurizer, 8 – PRZ injections, 9 – Reactorcore, 10 – Fuel assembly, 11 – Automatic control rod (ACR), fuel section, 12 – Automatic control rod (ACR), absorber section, 13 – ACR drives, 14 – Hydroaccumulators, 15 –- Spray system, 16 – Spray pump, 17 – Spray system tank, 18 – Low pressure emergency pump, 19 – LP emergency system tank, 20 – HP emergency pump, 21 – HP emergency system tank,22 – Containmentsuctionsump, 23 – Spray system cooler, 24 – Containment, 25 – Reinforcedconcretecontainmentwall,26 – Bubble condenser tower air trap, 27– Check valve, 28 – Bubble condenser tower, 29 – Bubble condenser towerflumes, 30 – HP stage of steam turbine, 31 – LP stage of steam turbine, 32 - Electricalgenerator, 33 – Unit transformer, 34 – Steamseparator and reheater, 35 – Condenser, 36 – Condensatepump, 38 – Condensatepump (stage 1), 37 – Condensatetreatment, 38 – Condensatepump (stage 2), 39 – LP regeneration, 40 – Feedwater tank, 41 – Main electricfeedwaterpump, 42 – HP regeneration, 43 – Coolingtower of circulating water, 44 – Circulating water pumps

Fig. 2: Technological diagram of NPP Dukovany

II.1.1.1.1 Equipment renewal program

An Equipment renewal program, called MORAVA (**MO**dernization - **R**econstruction - **A**nalyses - **VA**lidation), was approved in 1998 and realized in the following years. The program was followed by modernization actions realized within Completion of NPP Dukovany. One of the main goals of the Program Morava was to achieve the level of safety acceptable within the EU. Within the scope of the design, several actions were realized that are significant in view of safety, the so-called safety findings IAEA ("Safety issues and their ranking for VVER 440 model 213 Nuclear Power Plants") that were also assessed within the EU. Their realization is also connected with a significant decrease of probability value of reactor core melting. Significant safety modifications include:

- protection of CSS sumps – modification of screens as prevention of clogging of showering pumps suction with debris in the course of an accident,
- modification of the equipment on longitudinal auxiliary floor +14.7 m – protection of high-energy piping against flying objects,
- complete renewal of the control system (automatic devices of RTS, ESFAS, ELS, PAMS, etc.),
- relocation of the sectional collector of super-emergency feed pumps – separation of the safety system from high-energy piping against cold pressurization,
- completion of RPV emergency depressurization and SG primary circuits,
- drainage line from A, B 301/1,2 – prevention of loss of coolant in case of a leakage to the MCP board,
- modification for enhancement of fire protection – extension of stabile fire sprinkler system, installation of fire-resistant doors, fire extinguishing on the MCP board,
- modification of the design "leak before break" – higher resistance of the pressurizer body against vibrations, completion of equipment of circulation piping wipe restrainers,
- qualification of PSV and PRV for work with aqueous medium,
- qualification of SGSV and SBSA on steam piping for work with aqueous medium,
- realization of nozzles on SEFWP lines that enable connection of mobile LFRU pumps,
- replacement of EFWP for more powerful types,
- qualification of components important for safety,
- modification of geometry of the TL11 ventilation line in the containment to eliminate complete loss of coolant during LOCA. This action also includes preparation of an inflow opening for possibility of RPV external cooling,
- implementation of symptomatically oriented emergency operating procedures (EOPs) and instructions to manage severe accidents (SAMG).

II.1.1.1.2 Periodic Safety Review (PSR)

The PSR was carried out in JE Dukovany after 20 years of operation, in the period from 1/2006 to 6/2007. The process of PSR NPP Dukovany preparation was controlled so that general compliance with the instruction IAEA NS-G-2.10 and principles WENRA for PSR, defined in the document "Reactor Safety Reference Levels", were preserved. The PSR of NPP Dukovany was mostly performed by workers of ČEZ, a.s.; this complies with the mentioned instruction IAEA NS-G-2.10.

The purpose of PSR acc. to the instruction IAEA NS-G-2.10 was to support the thorough assessment of the condition of key fields influencing safety.

- determine to what extent the power plant meets the current internationally recognized safety standards and practices,
- verify validity of the licensed documentation,

- determine whether the corresponding measures are taken to maintain safety of the power plant by the next PSR,
- determine improvements in the field of safety that should be realized to resolve the identified safety deviations.

After executing the PSR, there was prepared, ahead of time, documentation that includes particularly Methodologies and Criteria based on legislative documents of the Czech Republic and documents IAEA up to the Safety Guide level, documents of the series INSAG and requirements of WENRA for PSR. Assessments were made for all fields (total 14 fields) and for all safety factors defined according to the instruction IAEA NS-G-2.10.

The found deviations were divided in view of safety significance and with use of the approved Methodology, and then divided into 4 groups (high, medium, low, very low).

The complex assessment within PSR identified similar opportunities to improve safety as are mentioned in this report.

Some of them have been realized or already almost finished (qualification of the design of NPP Dukovany incl. improvement of seismic resistance, completion of the system PAMS, creation of a seismic PSA, measures against irreversible loss of coolant, measurement of level in the reactor shaft, completion of equipment of the wipe restrainers, improvement of accuracy of containment tightness measurement, preparation of documentation for TSC, monitoring and evaluation of quality of the human factor and culture of safety), others are now in the phase of preparation for their implementation and would be realized regardless of this new assessment. In the field of management of ageing this includes implementation of a program on equipment lifespan and reliability control, in the field of technical issues this includes a program on improvement of resistance of the design of NPP Dukovany to consequences of severe accidents (that also includes increase of capacity of hydrogen liquidation system, completion of recombinators for severe accidents, preparation for flooding of the reactor shaft, installation of an inner seismic monitoring system, modification of the CR venting system and improvement of resistance or limitation of loading of the VBC 12th floor, modernization of the electronic fire alarm system, improvement of resistance of II.C. in the containment, improvement of resistance of covers – diesel sets and oxygen regeneration). In the field of administrative and personnel issues this includes completion of the creation of the Shutdown SAMG and creation of a means enabling graphic display of courses of severe accidents as a tool for creation, instruction and training of staff, completion of safety analyses of beyond design basis accidents and non-production states. PSR assumes implementation of approved measures by the year 2015, i.e. till the time of extension of NPP Dukovany service life.

II.1.1.1.3 Description of the main safety systems

Safety of JE Dukovany is ensured by the ability to fulfill these general safety criteria:

1. Shut down safely the reactor and hold it in conditions of safe shutdown.
2. Dissipate residual heat from the reactor core and from spent nuclear fuel.
3. Limit release of radioactive substances so that leakage does not exceed the fixed limits.

The systems perfoming the above safety functions are classified as systems important in view of nuclear safety which are further divided according to their function and importance into:

- Safety systems
- Safety related systems.

Systems important in view of nuclear safety, i.e. safety systems and safety related systems, belong to classified equipment and in compliance with legislative requirements are divided into three safety classes according to their importance in view of safety.

Technological systems, construction and components are also classified in view of seismic resistance. All safety systems (and some safety related systems) belong to the first category of seismic resistance.

Units of this type are characterized by their ability to ensure basic safety functions through the following multiple diversion systems in normal and abnormal operation modes and under emergency conditions:

- **Reactivity control** is provided by means of a mechanical regulation – by fall of control rods due to their deal load, high-pressure active safety systems with high concentration of boric acid, low-pressure active and passive safety systems with shut-down concentration of boric acid, safety related systems for refueling and boron control under normal and abnormal operating conditions.
- **Heat removal** is provided under normal and abnormal operating conditions by steam generators with a big stock of water; heat removal from turbine condensers is provided by the cooling water circulation circuit and cooling towers with natural draught. Bypass stations to the condenser, reduction stations, and technological condensers are designed for heat removal – all are parts of the safety related systems. To replace them, safety systems are also used– bypass stations to the atmosphere, possibly safety valves of steam generators. Cooling down of the unit is provided using a low-pressure system of cooling down with heat removal from technological condensers using ESW with redundancy 3 x 100% active components (pumps). Generally, the safety systems are designed as complete systems for each unit; essential technological water is designed for this type of power plant as systems for each double unit. Heat removal from ESW into the atmosphere is provided using cooling towers with spraying.
- **Elimination of release of fission products** from the reactor core is ensured by physical barriers – matrix and cladding of fuel, primary circuit pressure boundary, containment with maintained underpressure inside under normal and abnormal conditions. Heat removal from the spent fuel storage pool under normal and abnormal conditions is provided using a redundant (2 x 100%) safety related system; under emergency conditions it is provided through utilization of the stock of boric acid in the low-pressure safety system tanks, with evaporation into the reactor hall space. Under emergency conditions also activated is isolation of the containment from the surroundings with shutting off the fast acting valves at its border with heat removal and decreasing of pressure in the containment using an active safety spray system and heat removal using ESW into the cooling towers.

More detailed descriptions, including the method of solution of beyond design basis states, are given in the following chapters.

II.1.1.1.3.1 Reactivity control

Reactor core subcriticalty

The reactor active core consists of 349 assemblies, of this 312 working ones and 37 E&CR that may be displaced in a vertical direction. Spacing of the fuel grid ensures self-regulation properties of RC with generation of the power thanks to negative feedback from the power and temperature of the moderator and fuel (coefficients of reactivity from power and temperature are negative). Due to relatively small active core, it is stable against radial and axial variation of distribution of power with transition processes from xenon.

Two independent systems, based on different technical principles, are designed to control reactivity:

– Mechanical system of reactor shutdown including power supply switches

– Makeup and boron control system

Mechanical system of reactor shutdown belongs to the category of safety systems (SS). It consists of emergency and control rods (E&CR) and fulfills the following functions:

– ensures fast interruption of the chain reaction in the reactor by means of quick fall of the absorption part into the reactor core and at the same time by sliding out its fuel part from the reactor core,

– shares in automatic regulation with the purpose of maintaining the power of the reactor at the preset level and transition of one power level to another level,

– compensates fast changes of reactivity.

Total of 37 E&CR are located in the reactor; these are divided into six groups. E&CR fulfills its safety function of fast reactor shutdown by its fall due to its wight after switching off the power supply switches. Loss of power supply means automatic switching off the switches (fail safe).

Makeup and boron control system includes two basic operating systems, TK and TE.

TK (Technological condenser) system is designed above all for refilling of I.C (compensation of non-organized leakages). The system may also be used for refilling of I.C. in case of emergency situations (leakage of the primary circuit, rupture of the steam pipeline, rupture of a SG pipe), even if it is not designed particularly for these emergency situations. It is a technological circuit with pumping sets designed as 3 x 100% with power supply from secured power supply of category II (DG).

TE system (System for drainage of primary circle) is designed for drainage of coolant from I.C in all operating modes. It is in fact a technological circuit with working and backup pumps with power supply from secured power supply of category II (DG).

b) The boron control system includes two basic operating systems, TC and TB.

TC system (Continuous water purification system of the primary circuit). The system is designed, above all, to maintain the desired quality of I.C coolant at values specified by standards of the water-chemical regime.

TB system (Boric acid make-up system) is a system for refilling of the primary circuit with boric acid. The system consists of two storage tanks with H_3BO_3 concentrate (2 x 50 m^3) and six pumps (1 low-pressure pump (for manipulation), 2 high-pressure pumps (for pressure tightness tests), 3 emergency pumps, with power supply from secured power supply of category II (DG)).

In addition to the above described systems there are also designed core emergency cooling systems to ensure the safety function of reactivity control; these belong to SS.

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

a) TJ system – the high-pressure system of RC cooling is used to mitigate the course and liquidation of consequences of accidents connected with loss of I.C and/or II.C tightness. The equipment is in standby status under normal condition, ready to intervene automatically in case of occurrence of an emergency situation.

In case of an emergency situation the system ensures:

– refilling of I.C and increasing of boric acid concentration in I.C in case of leakage of I.C or rupture of II.C, with the purpose of reducing damage to fuel,
– prevents inadmissible transition processes connected with changes of reactivity,
– its function together with other safety systems reduces leakage of radioactive substances and penetration of ionizing radiation from the hermetic zone in case of arising of emergency conditions and afterward.

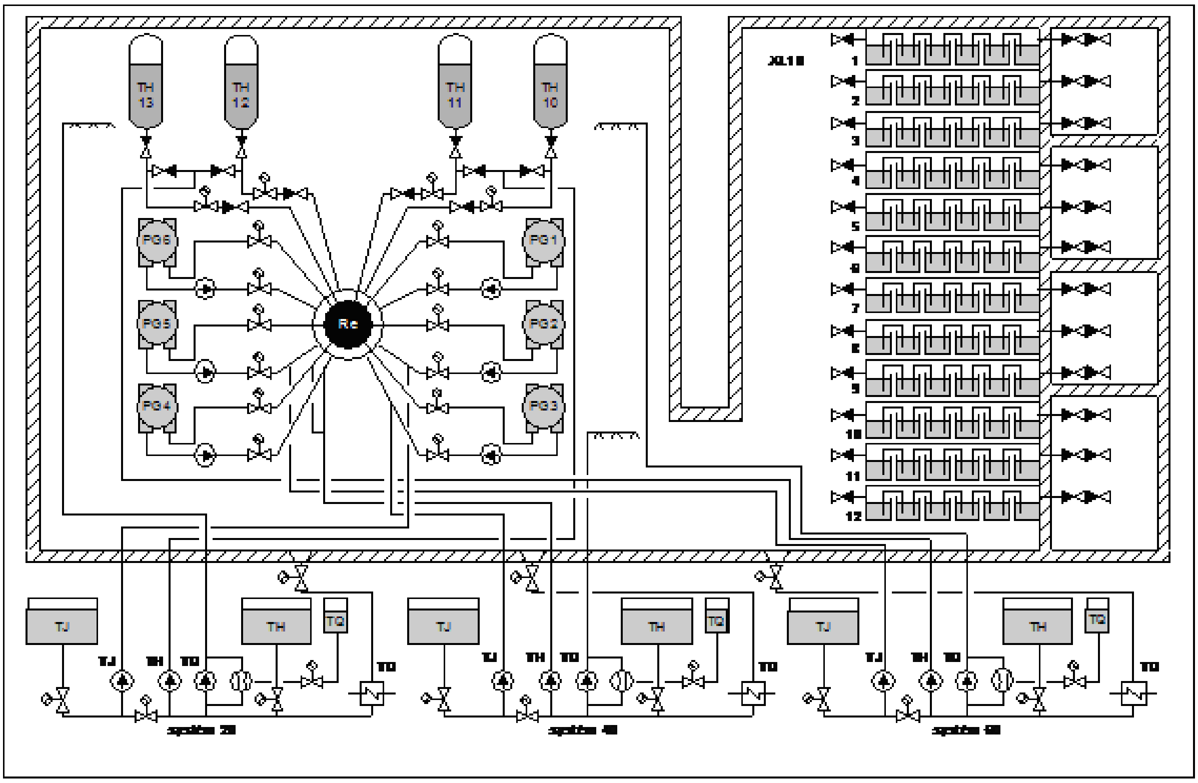


Fig. 3: Basic scheme of NPP Dukovany emergency systems

Re	Reactor pressure vessel
PG	Steam generators
TH	Hydro accumulators
XL10	Bubbler condenser
TJ	High pressure pumps and tanks
TH	Low pressure pumps and tanks
TQ	Spray pumps

The system is designed with a system redundancy 3 x 100% including all supporting systems (cooling, power supply, control and ventilation).

Power supply of TJ pumps is provided from SPSS category II. Main equipment of the emergency cooling TJ system:
– high-pressure emergency refilling pump - 3 pcs with flow rate 65 m^3/h and at the counterpressure 12.7 MPa; max. pressure at the discharge is 14.3 MPa,
– TJ system tanks - 3 pcs (3 x 80 m^3, concentration of H_3BO_3 40g/kg and temperature 50 °C).

b) TH system – the low-pressure emergency cooling system is used to mitigate the course and liquidation of consequences of accidents connected with large loss of coolant accident from I.C. The equipment is in standby status under normal operation of the unit at the nominal or reduced power, ready to intervene automatically in case of occurrence of an emergency situation. The TH system is drawn up similarly to the TJ system (i.e. with redundancy 3 x 100%).

Power supply of the TH pumps is provided from SPSS category II. Main equipment of the emergency cooling TH system:

– low-pressure refilling pump - 3 pcs - 280 m^3/h at 0.71 MPa at the discharge
– TH system tanks - 3 pcs (3 x 250 m^3, concentration of H_3BO_3 12 g/kg).

Emergency core cooling system TH - passive

The system of pressure storage tanks – hydroaccumulators - is designed to flood RC in the accident initial phase and ensures fast flooding of RC with boric acid solution. The system consists of 4 pressure storage tanks (4 x 40 $m^{3,}$ concentration of H_3BO_3 is 12 g/kg). Each of the four pressure storage tanks has its own independent discharge line DN 250 directly to the reactor. Discharging of the storage tanks to the reactor happens thanks to the nitrogen cushion passively by decrease of pressure in I.C.

Other TM, TD systems can also be used to ensure the reactivity control function in case of beyond design basis accidents in addition to the systems described above.

TM system – it is a system of purification of pool waters, belonging to SRS. The system is designed for purification of water in tanks with emergency stock of boric acid solution of the bubbler condenser, cleaning of the system of reactor core emergency cooling and spent fuel storage pool from soluble and insoluble additives. For the purpose of ensuring the reactivity control safety function it is possible to use unit pumps TM13,14 D01 of this system supplied from secured power supply of category II (DG) for repumping of boric acid solution between low-pressure emergency system tanks with its later delivery to the primary circuit after its depressurization.

Similarly the **system TD60** unit pumps can be used for repumping the concentrated solution of boric acid between the high-pressure emergency system tanks with its later delivery to the primary circuit system using high-pressure pumps of the emergency systems.

Subcriticality of the spent fuel storage pool

Subcriticality of the set of fuel assemblies in the storage pool is ensured by two independent systems:
- geometry and material design of storage grids installed in the spent fuel pool,
- concentration of boric acid in the pool volume.

As many as 699 spent fuel assemblies (FA) can be stored in the basic compact storage grid; of this, 17 for FA located in hermetic cases. In case of a shutdown with complete removal of fuel from the reactor, a so-called reserve storage grid is installed temporarily over the basic storage grid, at the level of +10.97 m. The reserve grid allows storage of 296 FA and 54 hermetic cases.

Minimal level +18.5 m is prescribed for storage of fuel in two layers and at least +20.75 m for manipulation with fuel.

Concentration of boric acid in water is kept at minimal value 12 g/l. According to the limit and conditions, temperature of water in spent fuel storage pool (SFSP) is prescribed $\leq$ 60 °C; according to the operating instructions (OIs), temperature of water shall be kept under 50 °C.

No limitations are valid for putting fuel into the basic compact storage grid. Rules for putting fuel in the upper, i.e. reserve, storage grid are prepared for putting FA so that FA with fresh fuel is surrounded with FA having a higher degree of burning out.

Under the above given conditions the subcriticality is ensured even in the case that the spent fuel pool is filled with pure condensate, i.e. $\mathbf{k_{ef} < 0.95}$.

II.1.1.1.3.2 Heat transfer from reactor to the ultimate heat sink

For the nuclear unit VVER-440, type 213 we can distinguish 7 operating modes of the unit.

Tab. 1: 7 operating modes of the unit VVER-440, type 213

MODE	MODE name	Reactor power [% N_{nom}]	Reactivity $\Delta k/k$ [%]	Temperature $T_{I.C}$, T_{RPV}, T_{HVS} [°C]	Pressure $p_{I.C}$ [MPa]
1	**Operation at power**	> 2	> -1	$T_{I.C} > 250$	$p_{I.C} > 8.3$
2	**Non-production operation** τ_R = 72 hours	≤ 2	≥ -1	$T_{I.C} > 190$	$p_{I.C} > 8.3$
3	**Hot standby**	Residual power	< -1	$T_{HVS} \geq 180$°C	$p_{I.C} > p_{atm}$
4	**Semi-hot standby**	Residual power	< -1	$T_{HVS} \geq 90$°C	$p_{I.C} > p_{atm}$
5	**Shutdown with I.O cooling down**	Residual power	< -1	$T_{HVS} < 90$°C	$p_{I.C} > p_{atm}$
6	**Shutdown with I.O unsealing**	Residual power	< -1	$T_{HVS} < 60$	$p_{I.C} = p_{atm}$
7	**Removal of fuel from** RC	RC **does not contain any fuel**			

T_{HVS} – temperature of I.O hot legs

All existing heat transfer means / chains from the reactor to the UHS under different reactor shutdown conditions

These chapters describe the ways of heat removal from RC to UHS in modes 3 to 6, i.e. the modes when the unit is shut down and it is necessary to ensure heat removal from I.C to the ultimate heat sink with possibility of unsealing of the I.C and dissipation of residual heat after it is unsealed.

Operating systems as well as safety systems can be used for heat removal for fulfillment of all the above mentioned reasons.

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

Operating systems

<u>Condensation system</u>

Normal operating system for heat removal from RC to the ultimate heat sink provides heat removal through the secondary circuit; heat removal from RC is ensured using forced circulation (if MCP are in operation) or using natural circulation. Heat is removed from I.C using SG, steam is led to TG. Heat is removed from condensers by condensate pumps that deliver water to the feeding tanks. Water is delivered from the feeding tanks to SG using normal or emergency SG feeding equipment.

Cooling down system

This system works in two operating modes, steam-water and water-water regimes. Heat removal is ensured in both modes by ESW in the technological condenser). Heat removal from RC to SG is ensured by force or naturally.

In the steam-water regime, steam is led from SG through the cooling down reduction station to technological condenser from which heat is removed by ESW. Condensate is led to collector tank, resp. FWT. The level of water in SG is controlled by the normal SG emergency feeding system. Any possible loss of water in the system is compensated by demineralized water pumps 1 MPa.

FWT and demineralized water tanks are used as sources of water, pumping is provided by electrical feed water pump or emergency feed water pump (EFWP) with power supply from SPSS category II. (DG); heat removal from technological condenser is provided by pumps ESW with power supply from SPSS category II; power supply of demineralized water pumps is provided by SPSS category II (DG).

All the mentioned systems (except ESW) belong to category SRS. ESW belongs to SS.

In the water-water regime, technological condenser is utilized as a heat exchanger water/water. Steam pipelines and main steam collector and whole cooling down system lines are filled with water using EFWP. The piping lines including suspensions are dimensioned for weight with water in piping. After complete filling of the lines with water, circulation of the coolant is provided by cooling down pumps. Any possible loss of water in the system is compensated from demineralised water tanks. It is also used at least one FWT that serves as a pressurizer.

All these systems belong to category SRS, demineralized water tanks 1,000 m^3 belong to category SS; technological condensers and pipelines are doubled (2 x 100 %), pumps are designed as 3 x 100%. The system is located in the machine room and not separated physically from normal operation systems. Feeding lines to SGs including regulation and isolation valves are installed on a longitudinal auxiliary floor at +14.7 m; steam pipelines from individual SGs including isolation valves are installed in the same area. Each of the technological condensers is cooled by another ESW division. The third ESW division is designed as backup equipment that can interconnect both technological condensers. The cooling down system is not seismic-resistant, only the technological condensers are designed as Sc, i.e. resistant against overturning. Protection against external events is the same as protection of all other systems installed in the machine room (see next chapter of this report).

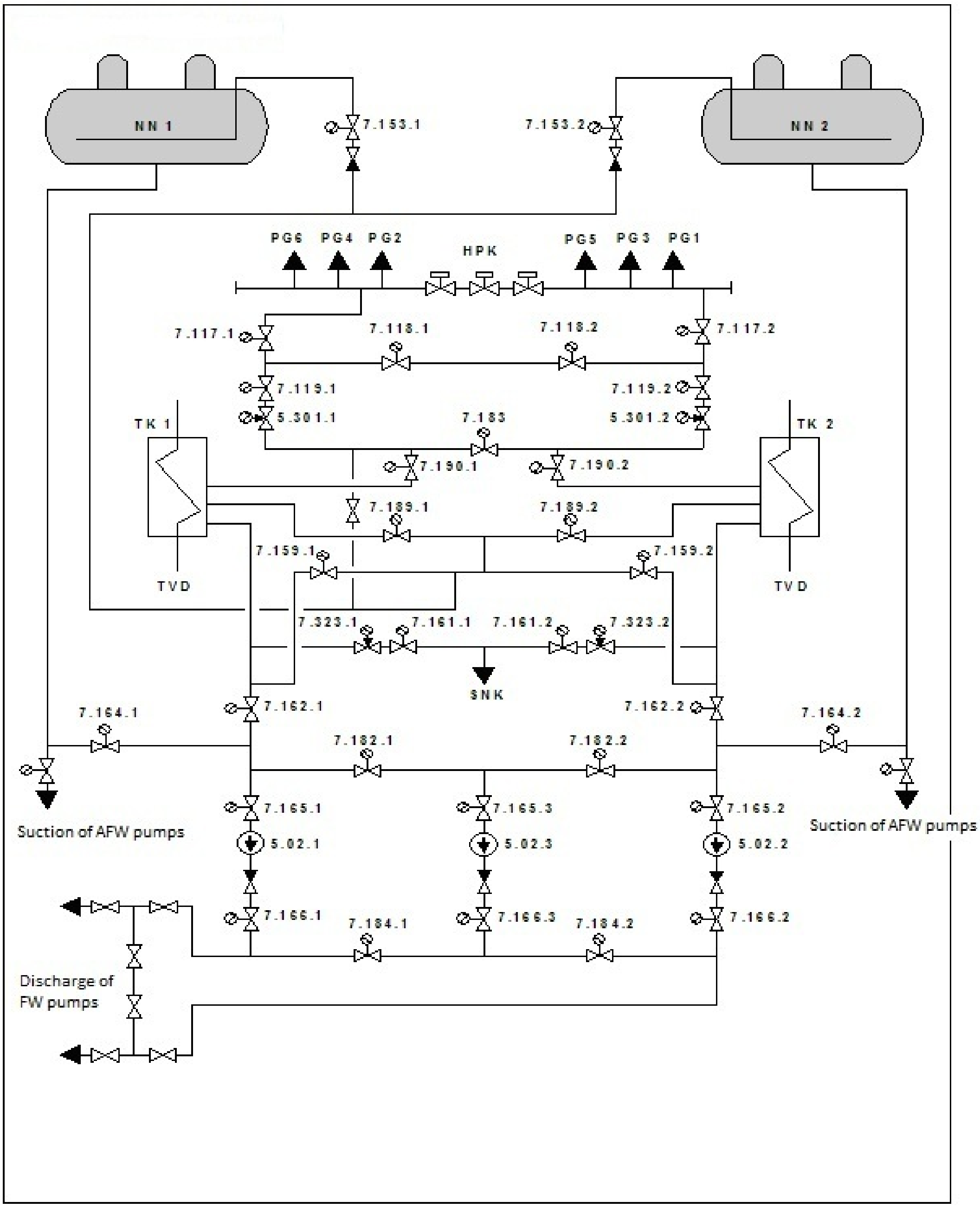


Fig. 4: Cooling down system of NPP Dukovany

NN1,2	Feed water tanks
HPK	Main steam collector
PG	Steam generators
TK	Technological condenser
TVD	Essential cooling water
5.02.1,2	RHR pumps

SG emergency feedwater system

The emergency feewater system consists of two emergency feed water pumps (EFWP) and doubled piping lines (2 x 100%). Suction is connected to the feeding water tanks (FWT), discharge piping lines are led outside the high-pressure heaters directly into the emergency feeding collector. Independent emergency feeding lines with a smaller diameter and with shut-off and regulation fittings are installed for feeding of SGs. These lines are interconnected with SG normal feeding lines and the connection to the containment and SG is made using joint piping for each SG. The system has secured power supply category II and the pumps are started using an ELS program. The system is installed in the machine room and not separated physically from normal operation systems; the lines are interconnected with the normal SG feeding system. The system is used for dissipation of residual heat using the technological condenser in the steam-water regime, for SG feeding at a low power (if TGs are shut down) and can also be used for SG feeding in the secondary feed&bleed mode with discharge of steam through SBSA or SGSV.

ESW is not necessary in the secondary feed&bleed mode, therefore long-term heat removal can be provided even in case of a loss of UHS. Power supply of the pumps and fittings is provided from SPS II; power supply of SBSA and SGSV is provided from SPS I, therefore heat removal can be realized even with TLHC, i.e. during operation of DG. However, in the case of operation of DG the ESW system will be necessary for cooling of DG. In the TLHC mode heat removal will be provided by means of natural circulation and the operators will proceed according to EOP regulations, procedure ES-0.2. By means of discharge of steam through SBSA the I.C can be cooled to approx. 120-130 °C in 30-40 hours after reactor shutdown. Dissipation of residual heat (without cooling down) within 0.5 hour after reactor shutdown can be realized by feed water with necessary flow rate to the SG 37 m^3/h, within 40 hours at approx. 10 m^3/h, in natural circulation mode at a rate10 °C/h and approx. 17 m^3/h within 40 hours. Delivery of feed water to SG is provided from FWT, refilling FWT from demineralized water tanks (3 x 1,000 m^3 for MPU) using demineralized water pumps 1 MPa. When proceeding according to ES-0.2, stock of demineralized water in the tanks 1,000 m^3, when taking into account simultaneous dissipation of residual heat from both reactors within one MPU, is sufficient for approx. 3 days; then volume of water in FWT tanks is available as well. The demineralized water tanks can be refilled from available stock of water in the NPP Dukovany site (sludge blanket tanks, cooling tower pools); stationary nozzles for easy connection of fire hoses are installed for this purpose.

The secondary feed&bleed mode can also be used in the water pipeline cooling down mode, i.e. after filling of SGs and steam pipelines with water; water would be discharged through SBSA or SGSV in this mode. Analyses are prepared for this cooling method, but operating rules are not prepared yet. This procedure requires a significantly higher flow rate to SG and therefore both EFWP or both SEFWP shall be used.

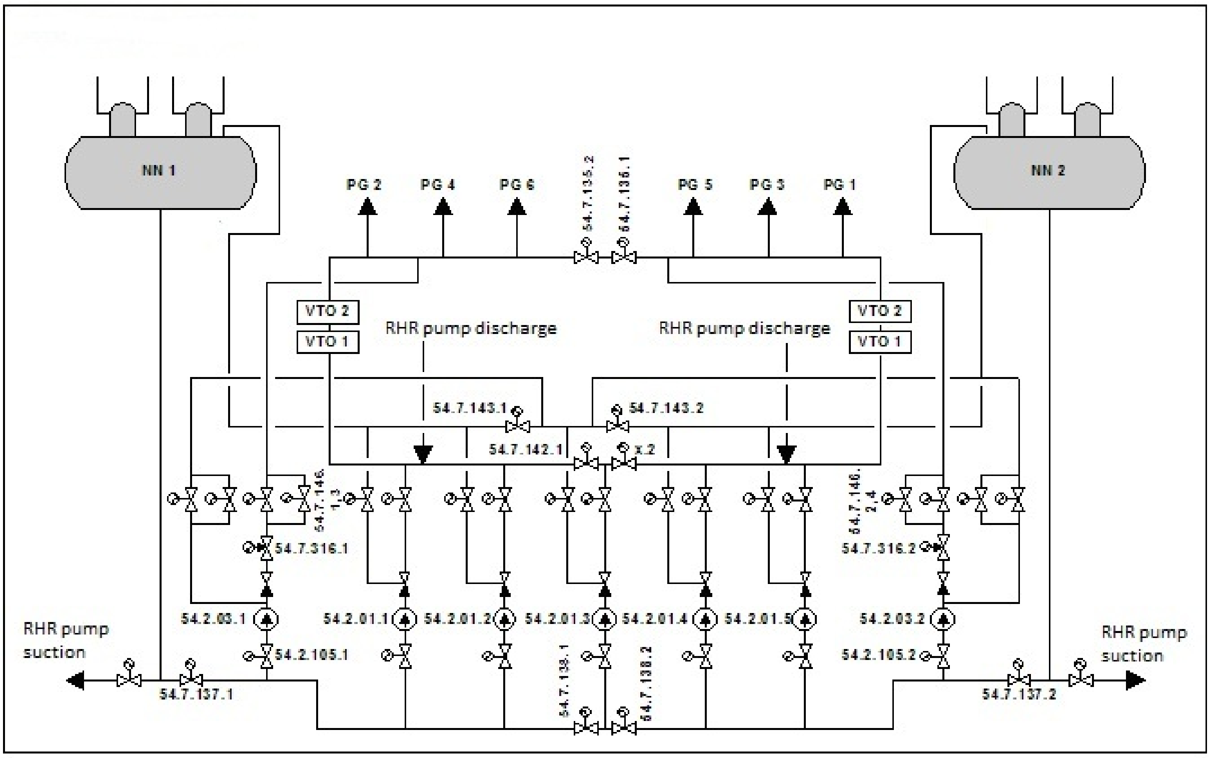


Fig. 5: NPP Dukovany SG feedwater systems

NN1,2	Feed water tanks
VTO	High pressure heaters
PG	Steam generators
54.2.01.1-5	Main feed water pumps
54.2.03.1,2	Emergency feed water pumps

In case of unavailability of the systems mentioned in the descriptions above, the systems belonging to category SS are available instead.

Super-emergency feeding system and SGSV & SBSA system

Heat removal from I.C is realized by means of forced or natural circulation. In this case, heat removal from II.C is realized through an unclosed circuit, i.e. through bypass stations to atmosphere or through SGSV to the atmosphere respectively.

Delivery of water to SG, if electrical feed water pump or emergency feeding EFWP are not available, can be provided using SEFWP pumps.

Three demineralized water tanks, installed in the area at the MPU wall, are sources of water for SEFWP. The tanks are shared by two units within the MPU. The temperature of water in the tanks is approx. 20 °C with protection against freezing in winter. SEFWP pumps are two for each unit and installed in a separate object outside the MPU, together with demineralized water pumps 1 MPa. The SEFWP pumps are installed in separate vaults and therefore separated physically. The SEFWP pumps and their lines are doubled (redundancy 2 x 100%). Power supply of the pumps and fittings is provided from SPS II; the pumps are activated by the ELS program as well as by ESFAS signals. The discharge lines are led in pipe channels to the machine room and continue to the floor +22 m on the longitudinal auxiliary floor. Feeding lines with shut-off and regulation fittings are separated physically; lines for SG 1,2,3 are led in one room and lines for SG 4,5,6 are led in another room. The lines penetrate the floor +14.7 m and through hermetic penetrations to the SG box to individual SGs. The super-emergency feeding pipelines are protected by a massive armoring in the machine room and longitudinal auxiliary floor for protection against flying objects. The SEFWP are separated completely from normal SG feeding lines and brought to SGs using separate nozzles.

The SEFWP system equipment is seismic-resistant, with minimal resistance to DBE-2; the SEFWP object and demineralized water tanks are resistant to seismicity as well as to effects of extreme climatic conditions.

SBSA and SGSV are located on the longitudinal auxiliary floor in the MPU at +147 m; SBSA are located on main steam collector as 2 x 100 % and a pair of SGSV is located on each SG. Power supply of SBSA and SGSV is provided from SPS I, remote control is possible from CR and from emergency control room; SBSA are provided with a small manual wheel to enable opening on the site. SBSA a SGSV are qualified to seismicity and HELB conditions; medium can be discharged in steam or water mode. SBSA enables reduction of pressure in II.C practically up to the atmospheric pressure. SG can be depressurized using SGSV only to 3.5 MPa. Currently a modification for optional reduction of pressure using SGSV up to the atmospheric pressure is being prepared.

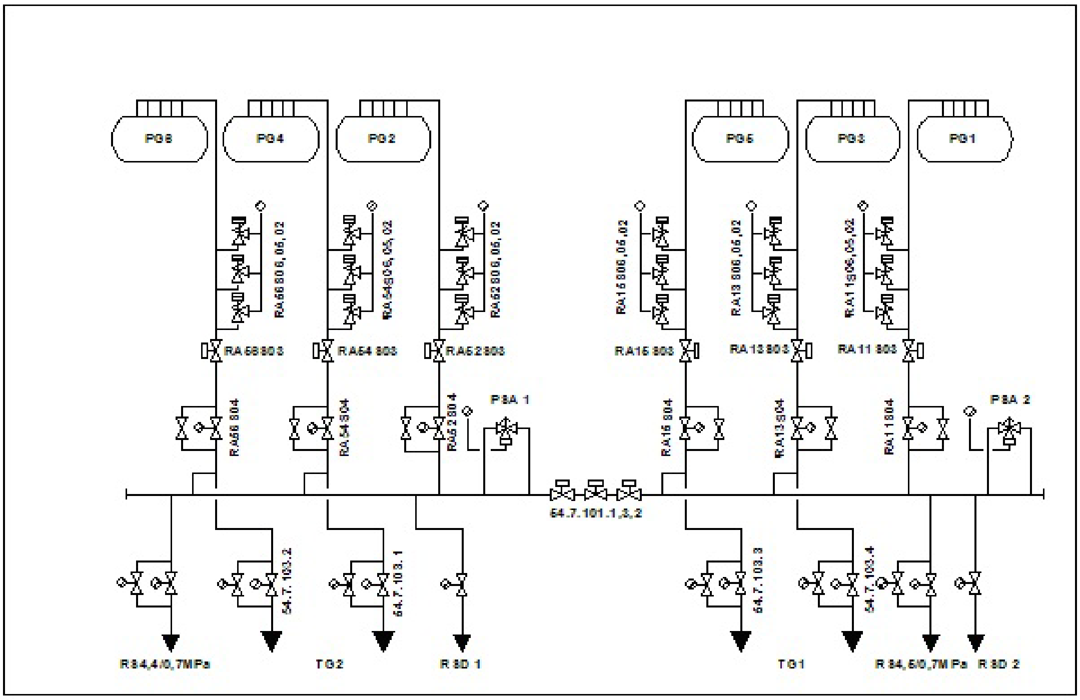


Fig. 6: Steamlines, PVSG, SBSA

PG	Steam generators
PSA	Steam bypass station to athmosphere (SBSA)
RA11,13,15,52,54,56 S02, S05, S06	SG safety valves

With use of the SEFWP system, demineralized water with temperature approx. 20 °C is delivered directly to a SG. A detailed procedure ES-0.6 was prepared within EOP for use of the SEFWP system. ESW is not needed for dissipation of residual heat; the secondary feed&bleed procedure can therefore be used with loss of UHS. However, if loss of UHS is connected with TLHC, ESW is needed for cooling of DG. With cooling down at a rate lower than 10 °C in the natural circulation mode the stock of demineralized water in the tanks 3 x 1,000 m^3 is sufficient for a period over 72 hours with simultaneous dissipation of residual heat from both reactors of one MPU. Demineralized tanks can be refilled from available stock of water in the NPP Dukovany site sludge blanket tanks, cooling tower pools using mobile equipment of the fire brigade; stationary nozzles for easy connection of fire hoses are installed for this purpose.

Cooling using ECCS

Another way to dissipate heat from I.C (except forced and natural circulation, considered in the previous descriptions) includes use of the high-pressure emergency system with TJ pumps in the feed&bleed mode, with discharging of coolant from I.C to the containment through PRV, resp. PSV. This mode provides heat removal from I.C to the containment and further through the ECCS exchanger using the ESW system. In addition, the TQ spray system would be used to keep pressure in the containment and condensation of steam. The high-pressure pumps can deliver a concentrated solution of boric acid to I.C even at normal pressure inside. The mode is analyzed with operation of one, two TJ pumps and opening of PRV or PSV and their combination. In the case that cooling of RC from the side of II.C cannot be restored, low-pressure pumps of the TH system can be used instead of TJ pumps after low pressure in I.C is achieved.

Power supply of all SS pumps is provided from SPSS category II. ECCS equipment is installed in two physically separated MPU system rooms.

Essential service water (ESW) systems

The ESW system is a key component in view of ensuring safety and transfer of residual heat, either from fuel in RC or from fuel in SFSP, to the ultimate heat sink.

With consideration of redundancy of ESW systems 3 x 100% and other inner redundancy 2 x 100% of each ESW division (4 pumps), loss of ability of heat transfer from the sources is conditional on the failure of all ESW pumps (total 12 pumps). In view of the physical separation of the systems and pumps, independence of power supply and other supporting systems, simultaneous failure of all ESW pumps is extremely improbable. Even in the case of operation of one pump in one ESW division only, fulfillment of basic safety functions can be ensured.

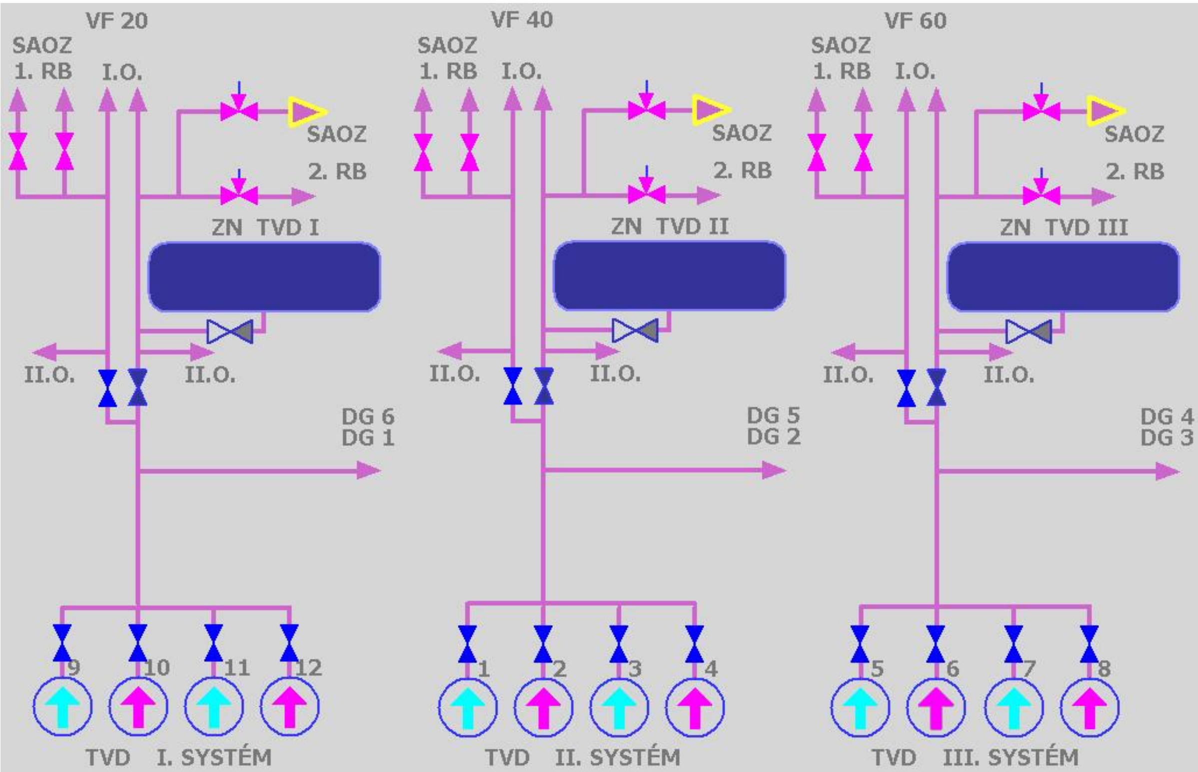


Fig. 7: ESW Systems of NPP Dukovany (note: in this scheme TVD = ESW)

Other systems

Because it is necessary to analyze all eventualities even in the field of beyond design basis accidents for heat removal from RC, also so-called alternative solutions can be utilized; these solutions are out of the scope of usual design solutions. This includes gravitational filling of SG directly from FWT without the use of refilling pumps. This mode could be used in combination with discharge of steam from SG through SBSA. The advantage of this arrangement is that it can also be used with SBO. It is sufficient to use SBSA for heat removal (power supply of these SBSA is provided from power supply sources of category I and can be opened even on site). For use of this method, it is necessary to depressurize SG to a pressure lower than the pressure in FWT (0.7 MPa).

Further, the cooling down mode can be utilized using EFWP resp. SEFWP and heat removal through SBSA. These systems allow so-called feed&bleed on the secondary side. This method enables cooling down at a slow rate to a temperature approx. 90 °C; it could be used for stabilization of temperature. One limitation is caused by the relatively low flow rate of the pumps used and limited capacity of tanks 3 × 1,000 m^3.

Alternatively the ESW of different systems can be interconnected on the technological condenser side.

II.1.1.1.3.2.1 Possible time constraints for availability of different heat transfer chains, and possibilities to extend the respective times by external measures (e.g., running out of water storage and possibilities to refill this storage).

Operating systems

It is assumed to use an cooling down system that is cooled by the ESW system. Heat removal by means of the condensation system and circulation water is not always available. In case of a conservative assumption of a fast shutdown of the reactor, heat removal from I.C will be ensured by discharge of steam from SG to the atmosphere using SBSA resp. SGSV and refilling of SG using EFWP from FWT (feed&bleed on II.C). Flow rate 30 t/h is sufficient for dissipation of residual heat. Loss of water in FWT is compensated from the demineralized water system 1 MPa. Warming-up of the cooling down system from cold state takes about 4 hours; this means loss of water in the demineralized water tanks approx. 120 tons. In view of their capacity of 1,000 m^3 this loss represents a negligible amount.

After changing to steam pipeline cooling using the cooling down system the circuit is closed and any loss of demineralized water is in fact eliminated. To enable the change to water pipeline cooling, it is necessary to fill up SG (6 x 24 = 144 tons) and steam piping lines (160 tons). Loss of water in FWT will be compensated again by refilling from demineralized water tanks 1,000 m^3. After the water pipeline circuit is started up, only volume changes of coolant will occur due to decreasing temperature.

Stock of water in FWT and stock of water in the demineralized water tank 1,000 m^3 is fully sufficient for cooling down of the unit as well as for continuous dissipation of residual heat in the water pipeline stage.

Safety systems

Super-emergency feeding system and SGSV & SBSA systems

In the case of cooling down of a unit in the regime feed&bleed on II.C (feeding of SG using the SEFWP pump and discharging of steam through SBSA resp. SGSV), the time of availability of this regime is given by the volume of water in the demineralized water tanks 1,000 m^3. With simultaneous cooling down of all units the volume of water in the demineralized water tanks and both FWT is sufficient for min. 4 days. In case of lower regimes (2, 3, 4) we can consider a longer time reserve. In regimes 5 and 6 this configuration is not considered because it is necessary to work in the steam regime to maximize availability of the regime.

Demineralized water tanks 1,000 m^3 can be refilled from fire trucks or using hoses from cooling towers, demineralised water station or retention tanks. This may extend the time of availability of the configuration with SEFWP for an unlimited period. In case of a failure of SEFWP, feeding of the SG can also be provided using FB pumps. Nozzles for connection of fire hoses are installed on the SEFWP building.

Cooling using ECCS

In the case of loss of heat removal from the II.C side, e.g. due to loss of feeding of a SG, heat removal from RC can be provided using the feed&bleed mode on I.C. This includes utilization of the high-pressure emergency system with TJ pumps in the feed&bleed mode with discharging of coolant from I.C to the containment through PRV, resp. PSV. In the case of a change to the recirculation stage a long-term heat removal can be realized because the cooling circuit is closed. Heat is removed through the TQ cooler to the ESW system. After low pressure in I.C is achieved, low-pressure pumps of the TH system can be used instead of TJ pumps. In the viewpoint of physics, the feed&bleed mode can be used in any regime (3, 4, 5, 6).

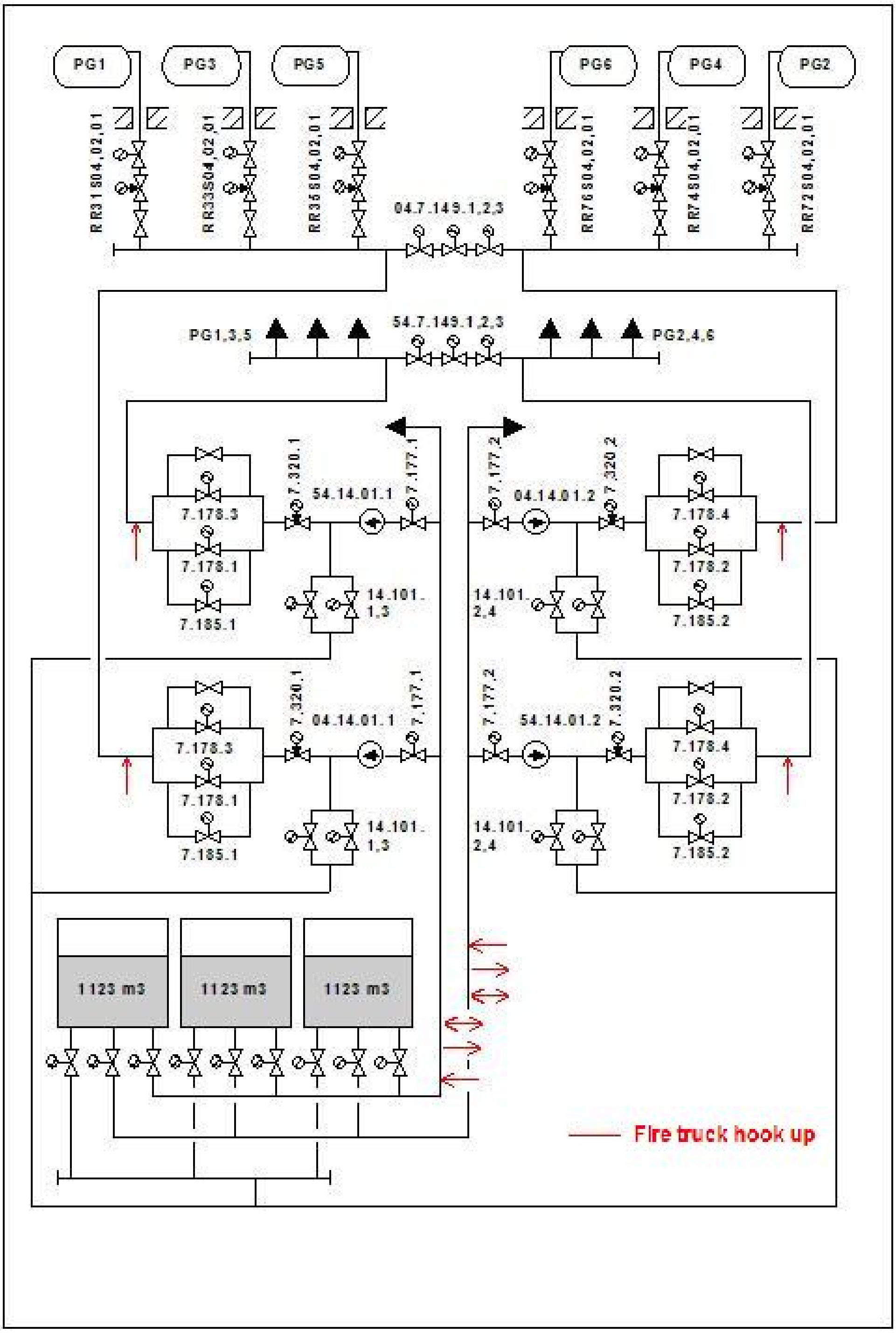


Fig. 8: Super emergency feed water system

PG	Steam generators
04.14.01.1,2 54.14.01.1,2	Super emergency feed water pumps

Other systems

Emergency feeding system and SGSV & SBSA systems

The feed&bleed mode on the II.C side can be operated even with use of EFWP pumps, whilst discharge of steam from SG is realized through SBSA and SGSV similarly as with use of SEFWP pumps. The only difference is suction of the EFWP pumps from FWT (SEFWP pumps with suction directly from demineralized water tanks 1,000 m^3). Drop of level in the FWT is compensated by work of demineralized water pumps 1 MPa with suction from demineralized water tanks 1 MPa. The disadvantage of this arrangement is that the temperature of water in the FWT is approx. 164 °C compared with demineralized water tanks 1,000 m^3, where temperature of water is approx. 20 °C to 30 °C. At the nominal level both FWT contain approx. 2 x 150 t of coolant. However, even in this regime it is possible to ensure heat removal from RC for a period of several tens of hours. More detailed specification of the period up to complete exhausting of the tanks could be made after some additional analyses.

The demineralized water tanks 1,000 m^3 can be refilled, similarly as in the regime with utilization of SEFWP pumps, using fire trucks or hoses from pools of the towers, demineralised water station or retention tanks. This can extend availability virtually for an unlimited period.

SBSA in combination with gravitational refilling of SG directly from FWT

In case of some SBO events it is possible to use FWT as a source for gravitational refilling of SG directly from FWT without use of any feeding pump. This means that it is a passive way of feeding of SG. Heat removal in this regime can be ensured for a period of approx. 20 hours after occurrence of SBO. This regime can also be used with a seismic event because the respective equipment is resistant up to the designed earthquake level.

II.1.1.1.3.3 AC power sources and batteries that could provide the necessary power to each chain (e.g. for driving of pumps and valves, for control of the systems operation).

Power supply of the components described in the two previous chapters is provided from power sources of categories I and II. SPSS category II also provides power supply to pumps, category I to SBSA and fast acting fittings as well as control systems.

II.1.1.1.3.4 Need and method of cooling equipment that belong to a specific heat transfer chain; special emphasis should be given to verifying true diversity of alternative heat transfer chains

In the final stage the cooling down, operating system provides heat removal using ESW. Steam pipeline and water pipeline systems can be considered as two physically different ways of cooling down because of the use of different pumps for their operation. Two technological condensers are available here, however, each of them is cooled by another ESW division; in case of a possible loss of one ESW division the third (backup) division can be connected to any of these two ESW.

Safety systems on II.C provide heat removal in the steam regime directly to the atmosphere (configuration SBSA plus cooling using SEFWP). In the case that feeding of this equipment in the operating mode is provided from DG, the ESW system is also necessary (for cooling of

DG). Nevertheless, this method of heat removal from the secondary side is the simplest method and can be realized with use of minimum means.

High pressure and low pressure ECCS always need ESW for heat removal. As mentioned above, ESW is necessary for cooling of DG in operation also in this case.

Necessity of ESW is applicable also in case of a configuration with discharging of steam through SBSA and refilling of feed water using.

Gravitational refilling of SG represents a passive way of heat removal into atmosphere; its functionality is limited by stock of water in FWT.

Refilling of SG using mobile equipment (fire truck) is the last considered option.

II.1.1.1.4 Heat transfer from spent fuel pool to the ultimate heat sink

In view of dissipation of residual heat from the spent fuel storage pool, two possible initial states can be distinguished:

- spent fuel from previous campaigns is put into the pool due to reduction of its activity and residual heat power. Minimum level in this state is at +14.6 m.
- during a period of extended shutdown of a unit due to general overhaul with removal of all fuel from RC, when together with spent fuel also partly spent fuel is also removed. Fuel in this state is arranged in two grids, one above the other and minimum level is at +18.5 m; in case of replacement of fuel it is at +20.75 m.

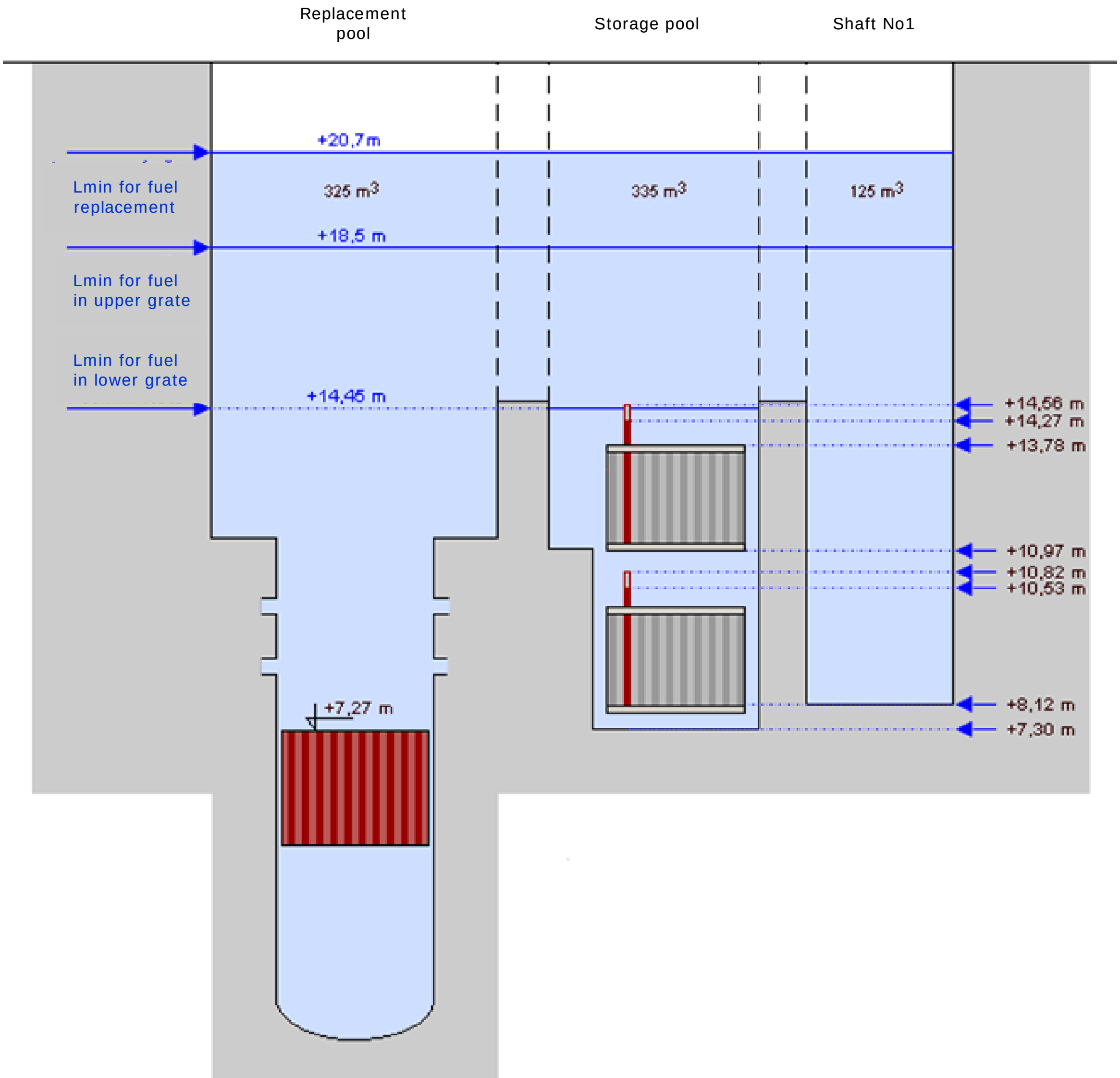


Fig. 9: NPP Dukovany Spent fuel pools

II.1.1.1.4.1 All existing heat transfer means / chains from the spent fuel pools to the heat sink

Heat removal from SFSP in both the above mentioned cases is provided by forced circulation of cooling water using the SFSP cooling system with marking "TG". Circulation is realized using two pool cooling pumps and heat is removed in two heat exchangers cooled by ESW. Simply said, two separate and independent cooling circuits exist (redundancy 2 x 100%). Due to an increase in the reliability and serviceability of the system of heat removal from refueling pool, both circuits are interconnected on the pumps suction side as well as the pumps discharge. This allows operational combinations in the chain of heat removal (pump, heat exchanger, ESW system). Power supply of the TG system pumps is provided from power sources of category II (DG) and the cooling system including the spent fuel pool itself is located in the MPU reactor hall outside the containment.

In the case that any of the forced cooling circuit combinations (pump, heat exchanger, ESW) is unable to provide heat removal from the pool, heat removal happens through boiling of water and evaporation of coolant from the pool to the reactor hall space where the power plant double unit is built (pools of both units of the respective MPU are just in the reactor hall).

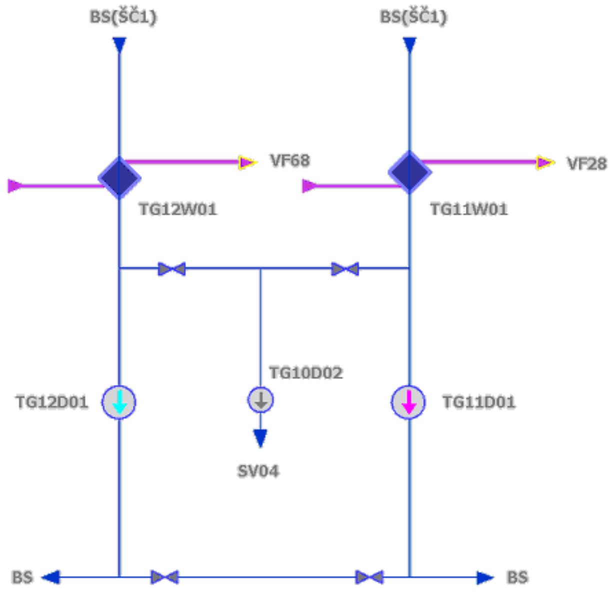


Fig. 10: NPP Dukovany diagram of TG

TG11,12D01	SFSP cooling pump
TG11,12W01	SFSP coooling heat exchanger

II.1.1.1.4.2 Respective information on lay out, physical protection, time constraints of use, power sources, and cooling of equipment

Operating systems

Heat removal under normal operating conditions is provided using two TG circuit systems that belong to SRS and are designed with redundancy 2 x 100%. Solution of boric acid is used as coolant. The cooling circuit is seismic-resistant. Verification calculations were made for the value PGA = 0.1 g in compliance with IAEA recommendations. The cooling circuit capacity is sufficient for both initial states of SFSP filling.

Dissipation of residual heat from SFSP is executed by continuous circulation of coolant through coolers that are cooled by essential technological water. ESW is the final recipient of residual heat under standard conditions. In the case of complete loss of removal of residual heat using the ESW system, this residual heat can be removed from SFSP for a long time by boiling of the coolant in the SFSP and refilling with a solution of boric acid, possibly with clean water.

Resistance to extreme external events

The SFSP cooling system is installed in the MPU reactor hall and therefore protected against adverse effects of extreme external events. Seismic resistance to the value 0.1 g was verified by calculations.

In the case of the inability of any combination pump-heat exchanger-ESW to remove residual heat from the fuel, heat removal can be provided by discharging coolant from SFSP to low-pressure tanks of the ECCS system with heating of water in these tanks. This does not cause any mixing of media because low-pressure ECCS tanks are designed for filling of SFSP in refueling regimes – solution from these tanks is used to increase the level in SFSP in the refueling regime. The pump TG10D01, designed for this purpose, can be used for discharging of coolant from SFSP. According to this procedure, it could proceed until the temperature in all ECCS tanks increases to 60 °C. This method can extend the time for reaching the boiling point of coolant in the pool. An operating instruction SD-9 is prepared for cases of insufficient cooling of SFSP.

As an alternative way of heat removal from the pool, also considered was the refilling of SFSP from VBC troughs using the pump VBC XL10D01 designed for filling of the troughs, possibly even without use of the pumps, but only by gravitational refilling.

Other possible way of refilling of SFSP is represented by use of pumps TM13(14)D01TM that are used for cleaning of coolant in SFSP; coolant can be delivered to SFSP using these pumps from tanks low-pressure ECCS. Power supply of the TM pumps is provided from system power sources of category II.

When the reactor is unsealed or in the refueling period there is also the possibility to deliver coolant using any pump of the high-pressure or low-pressure ECCS system directly to the reactor that is connected to the SFSP and from here to I.C, or delivery of coolant can be ensured by discharging from hydroaccumulators.

With use of coolant from all ECCS tanks and bubbler water trays the stock of coolant for compensation of losses due to boiling of coolant in SFSP is sufficient for more than 8 days, even in the case of the arrangement of fuel in two grids at two levels.

Delivery of water using mobile equipment with evaporation to the reactor hall is the last option of cooling down of SFSP.

II.1.1.1.5 Heat transfer from the reactor containment to the ultimate heat sink

The reactor and primary circuit are closed in a hermetic space called generally containment. The containment serves for the localization of radioactive (RA) substances in case of accidents connected with release of coolant inside the containment. The requirement for tightness of the containment is defined as max. 13% drop of weight of dry air within 24 hours at overpressure 150 kPa inside the containment. In the long term, tightness of containment of NPP Dukovany Units is 1.8 to 6.3%. The containment is equipped with a spray and vacuum bubbler condenser system that is used for reduction of pressure and localization of RA substances released to the containment with accidents connected with release of coolant.

Under normal operation of the reactor the containment is inaccessible for all operating staff and its tightness is checked periodically. All piping passing through the walls forming borders of the containment is provided with shut-off mechanisms that if an accident occurs:

- close – with systems for normal operation of the NPP, if these fittings were open before signal “severe accident”,
- open – with emergency and safety systems, if these fittings were closed before a signal “severe accident” that causes isolation of the containment.

The containment fulfils these following functions:

- limits release of radioactive substances outside localization of the accident,
- keeps radioactive radiation within limits specified by the design,
- protects the systems and components whose failure may lead to release of radioactive substances above the permissible limits.

The containment consists of these basic constructional elements:

- steel hermetic lining (liner)

• reinforced concrete protective structures

• hermetic closures

The total free volume (geometrical volume after deduction of large-volume technological equipment) of the containment for one unit (reactor) and without inclusion of volume of water in the bubbler trays is 51,119 m^3.

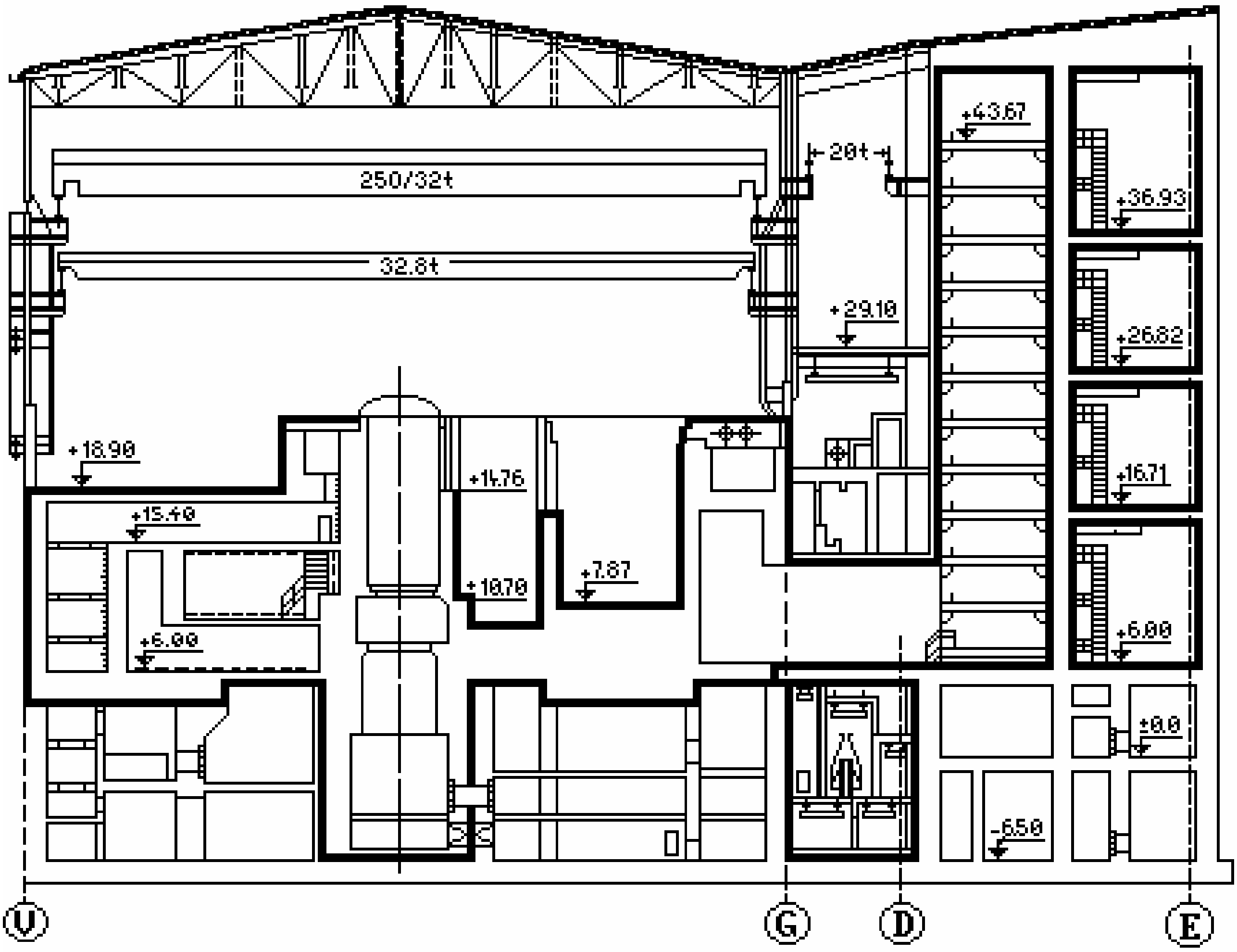


Fig. 11: Section through the NPP reactor hall with VVER-440/V213

II.1.1.1.5.1 All existing heat transfer means / chains from the containment to the heat sink.

Systems of heat removal from the containment can be divided into:

a) Venting recirculation systems (TL)

b) Spray system (TQ)

c) Vacuum bubbler condenser system (XL)

a) Venting recirculation systems (TL)

The concept of heat removal from the containment using venting systems is influenced by two requirements:

- It is necessary to release as low volume of air to the ventilation stack as possible to minimize values of activity released to the environment (air is carrier of activity).
- It is necessary to eliminate a large amount of heat released from the primary circuit (approx. 1 MW under normal operation) for maintaining a suitable environment inside the containment.

For these reasons venting recirculation systems are used for dissipation of released heat; these systems only circulate a large amount of air inside the containment and provide cooling and cleaning of air on special filters. Maintaining the desired underpressure is provided by smaller venting systems (inlet and outlet TL40, TL70) with exhaust of a small amount of air from the containment through a filtration system to the ventilation stack.

Venting recirculation systems are basic systems for heat removal from the containment in normal and abnormal operating modes and partly under emergency conditions. Circulation of air through heat exchangers ensures transfer of heat to the ESW system and therefore eliminates thermal load released by the primary circuit technology. Circulation also helps in elimination of consequences of a possible accident. The systems are installed in the containment, thus in the spaces to which a steam-gas mixture may spread (with accidents connected with loss of tightness of I.C, feedwater piping or main steam piping). The systems are dimensioned and designed to work under more difficult conditions in view of pressure, temperature and activity. Venting recirculation systems are installed in the ventilation centre that shares common air space with the steam generator boxes.

The reliability of operation of the venting recirculation systems is secured according to operating importance, with 50 to 200% backup units. Systems TL10, TL11, TL14 and TL13 are designed for heat removal from the containment. Heat removed from the containment is transferred by these systems to ESW or alternatively to the system of cooled water that further transfers heat to circulating cooling water.

The recirculation system TL10 is designed for the removal of excessive heat and moisture from the containment. It consists of three units – two working units and one backup unit. Power supply is provided from SPS category II.

The recirculation system TL11 is designed for the removal of excessive heat and moisture from the lower part of the reactor shaft and the reactor control and protection system shaft. It consists of three units – two working units and one backup unit. Power supply is provided from SPS category II.

The recirculation system TL14 is designed for the removal of excessive heat from the room of electric drives of the systems TL10 and TL11. It consists of three units – two working units and one backup unit. Power supply is provided from SPS category II.

The recirculation system TL13 is designed for the removal of excessive heat from the room of electric drives of MCP (MCP decks). It consists of two units – one working unit and one backup unit. Power supply is provided from a non-secured source.

b) Spray system TQ

The spray system is an active safety system designed to decrease pressure in the containment and for the heat removal that releases in the containment during accidents connected with release of coolant. Removal of heat is provided by means of heat exchangers of the ESW system. Each sub-system includes a spray pump with flow rate 380 to 520 m^3/h and overpressure 0.35 to 0.5 MPa at the discharge, an additional tank with capacity approx. 10 m^3 with solution of N_2H_4 + KOH, water stream pump, heat exchanger and a system of spray nozzles. Pressure is decreased by spraying cold water in the steam generator box space using spray nozzles. The system consists of three mutually independent circuits; each of the circuits is sufficient to manage all the states assumed by the design. The TQ system suction is connected to the tanks of the TH system (3 x 250 m^3 with concentration of H_3BO_3 12 g/kg) and after the tanks are emptied, the system converts suction from floor of the SG box through coolers transferring heat to ESW. Power supply is provided from SPS category II.

c) Vacuum bubbler condenser system XL

The vacuum bubbler condenser system is a passive safety system designed for reduction of initial growth of pressure in the containment in the case of an accident connected with release of coolant. Its function is to absorb a significant part of thermal energy released from the primary circuit coolant or released medium from the part of the secondary circuit that is located in the containment (feed water, steam). The vacuum bubbler condenser system consists of a set of water trays arranged in 12 levels – floors and filled with a solution of boric acid in concentration of 12 g/kg that absorbs the released heat. Volume of one tray is approx. 114 m^3. The trays form a water closure with low hydraulic resistance. The water closure lower space is connected to the containment; the space above the closure is connected to retaining chambers – air traps through a check valve DN 500 and then connected back to the containment through check valves DN 250. One air trap is shared by three adjacent flors of water trays, thus there are in total 4 aitr traps with total volume of 17,080 m^3. The air traps are designed for retaining non-condensed gases. The whole vacuum bubbler condenser system is part of the containment. In the case of occurrence of leakage on I.C an air-steam mixture is generated in the containment and flows through the water trays. The steam phase condensates in the water closures. Air continues to the retaining air traps where it causes automatic locking of the check valves DN 250. Activation of the spray system causes reduction of pressure in the containment and overpressure in the air cushion over the level of the water trays pulls out water from the trays on the steam generator floor through the spreading screens – this represents passive spraying of the containment. Power supply or instrumentation & control system is not required for function of the system as the system functions as a passive system.

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

The spray system is installed in the reactor hall outside the HZ and discharge pipes run to the containment. The spray system pumps are installed on the floor -6.50 m in the room of emergency systems 001, 002, 003. The spray system includes a system redundancy 3 x 100% including all supporting systems (cooling, power supply, control and ventilation). Under normal operational conditions of the unit at temperatures of the I.C coolant above 90 °C the spray system equipment in standby mode is ready to intervene automatically in the case of occurrence of an emergency situation when it receives a signal “overpressure in the box”. Its power supply is provided from SPS category II. The systems are designed for long-term operation. Cooling of the TQ system room is provided by its own venting system TL22.

The vacuum bubbler condenser system including air traps is located in the containment reactor shaft. The vacuum bubbler condenser water trays are located in the shaft of localization of accidents that is connected with the SG box by means of a corridor with cross-section 77 m^2. The system is permanently ready to fulfill its designed function during operation of the unit. It is designed as a passive system, without any need for power supply or control. Cooling is provided by the system TL10.

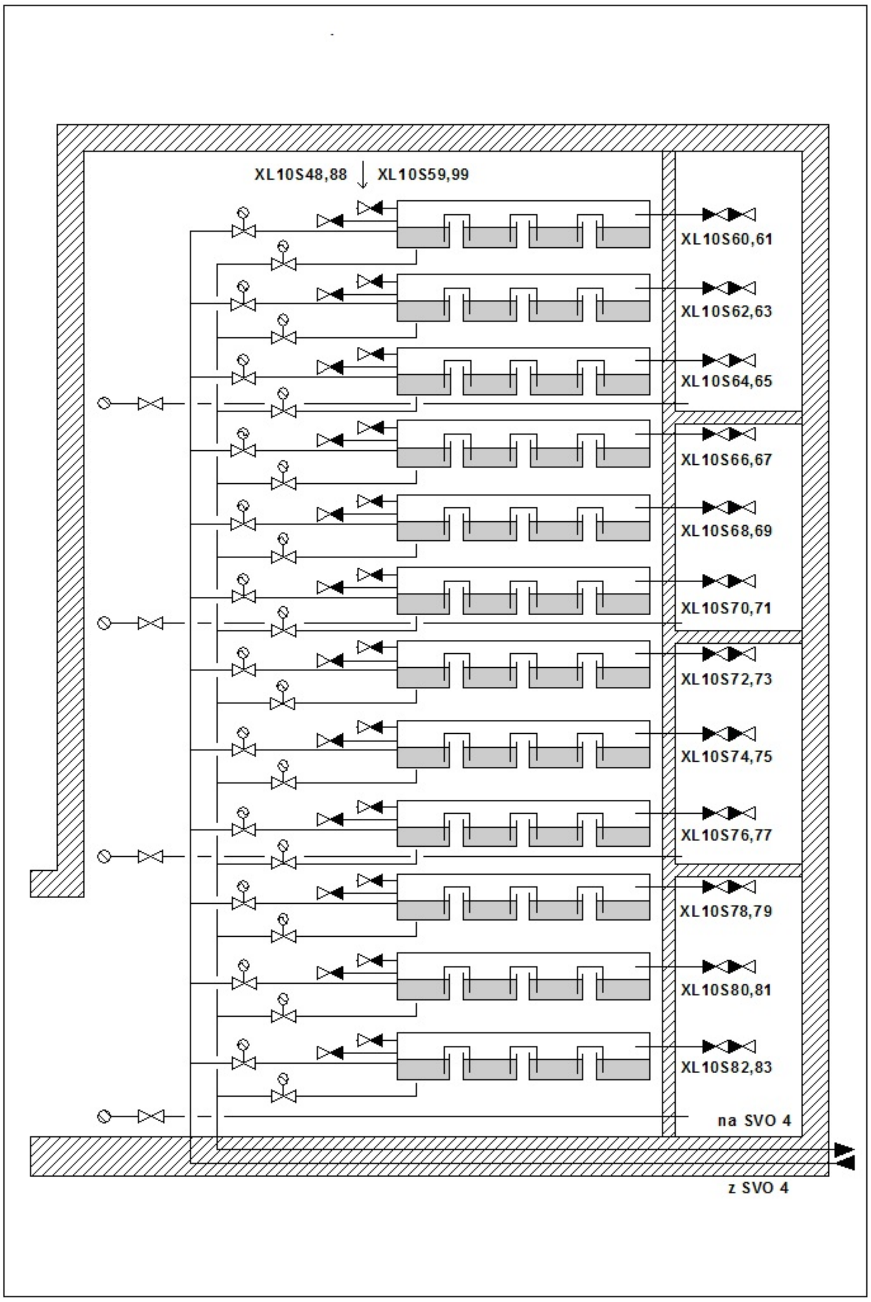


Fig. 12: Bubbler condenser

II.1.1.1.6 AC power supply

The nuclear power plant Dukovany (NPP Dukovany) consists of four reactor units VVER 440 MWe. The electrical system is designed so that its work is based on a principle of depth protection and ensures reliable power supply to all systems in view of nuclear safety in all operating, abnormal and emergency states.

The principle "depth protection of electric systems" (DIDELSYS) is applied on the electrical systems of NPP Dukovany in connection to solutions of the mechanical part and I&C instrumentation. Application of DIDELSYS begins in external high-voltage networks to which NPP Dukovany is connected and continues through systems of normal power supply of house consumption up to autonomous systems of secured power supply. Functions and linkages of individual DIDELSYS levels are shown briefly in the following table.

Tab. 2: The levels of depth protection of electric systems in NPP Dukovany

Level	Whole NPP (DID)	NPP electrical systems (DIDELSYS)	Robustness of levels
1	Prevention of deviations from normal operation	• Insensitivity to deviations U,f • Stability of transfer of energy • Dynamic stability • Island operation	• Linkage to robustness of technology, I&C and construction • Robustness of electrical systems (independence, redundancy, diversity) • Protections • Regulation, automatics • Quality • Testing of the function • Operating instructions • Trained staff
2	Identification and remedy of events, states and conditions of abnormal operation	• Regulation of TG on house consumption • BES to backup power supply	
3	Interventions (measures) leading to warding off, development or managing emergency conditions by design means	• Design (safety) functions of secured power supply: • SS (1,2,3), • SRS (4,5)	
4	Prevention and mitigation of consequences of extended design conditions	• Procedures for managing SBO • Measures to support mitigation of consequences of SA • (Function AAC)	
5	Measures of protection with a radiation accident	Support of emergency control centres	

The structure of DID and the robustness of individual levels create resistance to external and internal events (failures).

Power supply of important NPP Dukovany protective and control systems and the executive systems that fulfill safety functions is provided from redundant systems of secured power supply (SPSS). Each NPP Dukovany unit has three independent safety SPSS (marked 1, 2, 3) and other SPSS classified as related to safety (marked 4.1 and 4.2). These SPSS provide supporting safety functions such as secured power supply and participate in control of functions of electricity consumers.

II.1.1.1.6.1 Electrical system outside NPP Dukovany

The nuclear power plant Dukovany (is located in the southeastern part of the Czech Republic. Output power of individual reactor units is increased gradually from 440 MWe to

500 MWe within the scope of the design V261. Pairs of reactor units are arranged in two MPU. Outlets of the NPP power and grids of operating power supply are made as a unit grid, the grid of backup power supply from the 110 kV network is made as a double-unit grid. Output power is led to the 400 kV distribution system (operated by ČEPS a.s.). Backup power supply is ensured from the 110 kV distribution system (operated by E.ON).

Information on reliability of the external electrical system

No fault in the networks 400 kV and 110 kV that might show any unsatisfactory function or reliability of the external system or incorrect reaction of NPP Dukovany to faults in the external network was recorded during operation of NPP Dukovany.

Some faults of the equipment occurred in connections of the units to the network or on the surrounding switchyards and transmission network lines during operation of the units. A summary of these can be found in the table below. A short-circuit on busbars of the 400 kV switchyard Slavětice in 1990 was the most serious fault and disabled connections of all 4 units to 400 kV and 100 kV networks. On the other hand, the system faults in the transmission system in 2006 demonstrated the ability of NPP Dukovany units to manage island regimes of the network operation.

Reliability of connection of NPP Dukovany to the external electrical networks and resistance of NPP Dukovany to faults is based in this field on the following properties:

- Block arrangement of the scheme on outlets of the output power. This arrangement limits transfer and spreading of faults among the units to each other. In combination with the robust scheme of the switchyard Slavětice (2 switches per branch, sectional division of busbars, and selective system of protections), it limits transfer of faults between the units and transmission system. The outlet of the NPP Dukovany output power to the network is designed according to the reliability criterion N-2.
- High functional and physical independence of the system that provides outlet of the output power 400 kV (i.e. working power supply of house consumption) and the system of backup power supply of house consumption (110 kV). The option to supply the backup power supply system from various geographically and functionally different sources.
- Reaction of the unit to faults and transition processes in an external network are controlled by a set of regulation devices, automatic systems and protections. These functions are mutually coordinated to ensure mutual selectivity and controlled decreasing of the unit output power per individual levels of depth protection as necessary.
- Static stability of transfer of the power to the system. The NPP Dukovany units are usually incorporated into a system of automatic secondary regulation of voltage and reactive power. This system ensured stable voltage in the pilot node Slavětice.
- Stability of the turbine set in case of short-circuits in the system of output power outlet (fast basic and backup protections that disconnect the short-circuit, effective regulation of the turbine and voltage of the generator). Stability was analyzed on a dynamic model of the transmission system. With activation of the basic and backup protections (up to 100 ms) and automatic devices in case of a failure of the switch 400 kV the turbine sets are stable when considering the function of their regulation.
- The ability of the units to be operated in an island regime of the transmission system that is characterized by large frequency and voltage variations (support of stability of the network with system faults). The units can be operated at full output power within the range 48.5 to 50.5 Hz. Operation of the units, limited in time and power, is possible within the range 47.5 to 52.5 Hz. The units are equipped with network frequency

protections that switch over regulation in v 1° (± 200 mH) of output power of the unit to an "island regulation". The unit also regulates its output power so that it supports to stabilize the conditions of U and f in the island network. In the case that the conditions cannot be stabilized and frequency variations still increase and exceed the limits (47.9 Hz for a period of 1 second or 52.5 Hz for a period longer than 10 seconds), the unit is disconnected 2° of frequency protection from the network and regulates itself to house consumption.

- Disconnection of the unit from the 400 kV network in the case of a voltage drop on 6 kV switchboards is provided by house automatic device that intervenes with 0.7 Un/1.5 s. After disconnection from the network the unit regulates itself to house consumption. In the case that regulation is unsuccessful, its house consumption power supply is converted to backup sources (network 110 kV). In the case of unsuccessful conversion to the backup sources (network 110 kV) or unavailability of these sources, power supply is converted to emergency sources (DG).
- The main unit switchboards are equipped with automatic switching from working to backup power supply (110 kV). In the case of a loss of the working power supply of house consumption (e.g. intervention of protections of the generators, unit transformers and other equipment that ensures outlet of the output power, or in the case of unsuccessful regulation of the turbine set) the switchboards are switched over to the backup power supply. In the case that switching over is unsuccessful, the secured power supply systems are eliminated and emergency sources (DG, battery system) are activated.
- Power supply of regulation, automatic and protective devices is provided from sources secured by batteries. Their functions are therefore independent of voltage drops in the network due to failure. A principle of electromagnetic compatibility is applied in the whole NPP Dukovany design so that functioning of the systems in the given electromagnetic environment as well as in the case of interference.
- The NPP Dukovany units are operated in compliance with the dispatching control of the transmission system. The transmission system operator knows the properties and operating limits of NPP Dukovany and this information is incorporated into the FA Code. Periodic maintenance of equipment in external networks (switchyards, lines transformation 400/110 kV) and equipment of NPP Dukovany is carried out in mutually agreed ways. In the case of emergency situations in the electrification system (disintegration of the network, SBO NPP Dukovany, etc.) restoration of power supply of NPP Dukovany house consumption from the transmission system has the highest priority.

II.1.1.1.6.2 Connection of the power plant to the electrification system

The basic electrical system of NPP Dukovany is shown on Fig. 13. It includes Units 1 & 2; the system of Units 3 & 4 is analogical. The grid also includes outlets of electrical power to a 400 kV network, backup power supply from a 110 kV network and the system of power supply of house consumption up to the level of main switchboards 6 kV and wiring of SPSS 1, 2, 3.

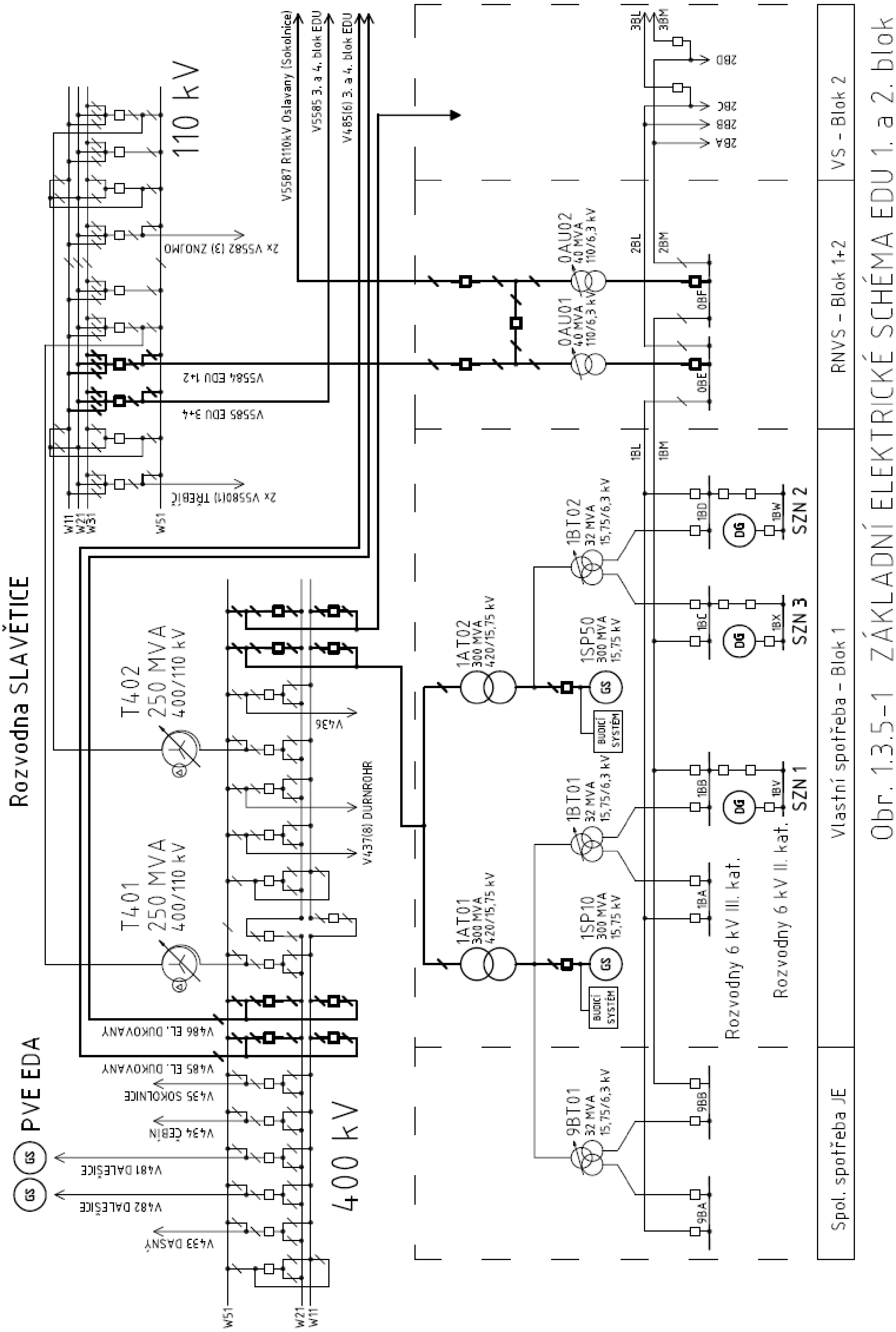


Fig. 13: The basic electrical system of NPP Dukovany – Switchyard Slavětice

Each reactor unit is equipped with two turbine sets with increased power and generators 300 MVA, 15.75 kV for realization of the design on the utilization of design reserves. Output power of each generator is led through a generator switch and unit transformer (300 MVA, 420/15.75 kV) to a power plant switchyard 400 kV. In this switchyard the outlets of both generators are connected together and led using a simple line to the 400 kV switchyard Slavětice at a distance of approx. 3 km from NPP Dukovany. The power plant switchyard only includes disconnectors, instrumental transformers and lightning arresters. 400 kV switches of the NPP Dukovany units are installed in the switchyard Slavětice.

The 400 kV switchyard Slavětice is connected to the transmission system using 6 lines; 4 lines distribute the output power to various distant parts of the Czech Republic and 2 lines to Austria. This creates a geographical diversification in the 400 kV wiring. Some of the lines are simple lines, others are doubled. The Czech transmission system as a whole is designed and operated in compliance with the criterion N-1. However, outlets of power from the switchyard Slavětice to the network are designed according to a stricter criterion N-2. These requirements are specified in the FA Code.

The switchyard 400 kV Slavětice is designed for outdoor operation with short-circuit resistance 50/125 kA. Its diagram includes two busbar systems and an auxiliary busbar. The main busbars are split in longitudinal direction by switches. The NPP Dukovany units are connected by wiring with two switches per branch; it is chosen in compliance with increased requirements for reliability and resistance of outlets of the NPP Dukovany output power to faults. Other outlets (to 400 kV lines of the transmission system are equipped with one switch per branch. The switchyard includes two transformers 400/110 kV, 350 MVA to provide power supply of the switchyard 110 kV Slavětice. Further units of pumped-storage power plant Dalešice (4 x 112.5 MW) that represent a backup AAC source for solution of SBO on NPP Dukovany are connected to the switchyard using two 400 kV lines.

Backup power supply of NPP Dukovany is provided from the 100 kV network. Each MPU has its own backup source with power supply provided from two different switchyards in the 100kV distribution network. Automatic switching over is installed between both supplies from the 110 kV network. These sources are connected usually to 110 kV switchyards Slavětice and Sokolnice, where transformations 400/110 kV resp. 220/110 kV are available. Also other power supply options exist. This ensures a wide variability and geographic diversity. This wiring also provides backup sources for transformers 400/110 kV and 220/110 as sources of backup power supply of NPP Dukovany with preservation of sufficient short-circuit hardness of voltage for backup power supply of NPP house consumption.

The 110 kV switchyard Slavětice is mainly used as the main source for backup power supply of the NPP Dukovany units. In addition, the 110 kV distribution network in southern Bohemia is also supplied from the switchyard. The 110 kV switchyard Slavětice has a robust and flexible wiring with three busbar systems.

The switchyard Sokolnice as the other usual source of backup power supply of NPP Dukovany is a significant junction of the transmission system with transformations 440/110 kV and 220/110 kV. It is located about 50 km from NPP Dukovany.

Wiring of switchyards 400 kV and 110 kV Slavětice, Sokolnice (and other substations in the surrounding of NPP Dukovany) and their operating methods are chosen to eliminate as much as possible any transfer of faults among NPP Dukovany units and between the NPP Dukovany units and the electrification system.

The NPP Dukovany backup power supply system can therefore be supplied from different geographically and directionally diversified sources in the transmission system (transformations 400/110 kV a 220/110 kV Slavětice, Sokolnice and Čebín). The hydro-electric power plant Dalešice was chosen as the main external AAC source after executing SBO analyses and this function was verified practically by tests. The power plant Dalešice (output power 4 x 112.5 MW) has the ability of a black start. The test verified the ability to provide power supply within 30 minutes (through the 400 kV line) or within 60 minutes (through the 110 kV line). The test included verification of technical and organizational measures to manage SBO, functionality of the communication means, roles and procedures of key persons in case of occurrence of SBO. Use of the hydro-electric power plant Dalešice, which is 6 km from NPP Dukovany, is conditioned by serviceability of the switchyards and 400 kV and 110 kV lines in the power supply line.

The test also verified the ability to provide power supply within 60 minutes (through the 110 kV line) from the hydro-electric power plant Vranov.

II.1.1.1.6.3 Electrical system inside NPP Dukovany

Main power supply sources and distribution networks

Power supply of house consumption of the consumers is distributed into several switchyards, supply systems and sources that are provided with a backup (on substitutive or redundant principles). This limits consequences of failures of these systems for operation of the reactor and unit.

Electricity consumers are divided into groups according to their importance and their power supply from the sources and networks of the respective category of secured power supply. The importance of the consumer includes a criterion of the consumer (safety) function and permitted time of interruption of its power supply. The consumer function is classified according to IAEA standards as safety function (SS), function related to safety (SRS) and function not important in view of safety (NSRS). House consumption of each of the NPP Dukovany units has the following sources available:

- Working sources, i.e. house transformers with voltage regulation (with power supply provided from turbine generators 300 MVA (259 MVA in case of units before running reconstruction) and/or from the 400 kV network). The working sources are characterized as NPP unit sources.
- Backup sources, i.e. reserve transformers with voltage regulation (with power supply provided from the 110 kV line). There are two reserve transformers for each reactor twin unit (MPU) and may be connected with reserve transformersof the neighboring MPU. The backup transformers can provide shutdown of one reactor unit in the case of a loss of its working power supply, with preliminary loading by consumers of the other reactor unit.
- Emergency sources that provide power supply to secured power supply systems (SPSS). The emergency sources consist of diesel sets, batteries and machine sets of non-interrupted power supply (rectifiers, inverters). These sources are installed in the NPP Dukovany site, dimensioned according to requirements of the supplied loads and their functional ability does not depend on conditions of working and backup sources or external network. Each of the NPP Dukovany units is equipped with three redundant SPSS (marked 1, 2, 3) classified as SS (each of them represents a supporting system for its SS division). SPSS 1 and 2 provide power supply of SPSS 4.1 and 4.2 for power supply of SRS and NSRS consumers.

The consumers not important in view of nuclear safety (NSRS, ensuring operation of the unit, production of electrical energy, etc.) are supplied from working sources. In case of a loss of working power supply (house transformers) this power supply is converted automatically to backup sources (reserve transformers). Each NPP unit includes 4 main units 6 kV switchboards (BA, BB, BC, BD) equipped with BES for backup power supply.

The consumers important in view of nuclear safety (SS and SRS) are supplied from secured power supply systems (SPSS). These SPSS consist of secured power supply networks and emergency sources. SPSS are usually supplied from working or backup sources (through BA, BB… switchboards). In the case of a loss of this power supply, the respective SPSS is disconnected from the normal power supply network and connected to power supply from emergency sources. Non-interrupted power supply of sensitive consumers is ensured by batteries.

Disconnection of SPSS and start of a DG is initiated by a loss of power supply (U < 0.25 Un for a period 2.5 sec or 0.7 Un/6 sec or by a drop of frequency to 47 Hz/6 sec). Variations of frequency are eliminated by a network frequency protection that evaluates drops of frequency in the 400 kV network. Design analyses and tests confirmed selectivity of this adjustment against the BES function from working to backup sources and function of other protections and automatic devices.

These initiating conditions mean that DG are started and loaded gradually by a fixed ELS program. According to safety requirements the DG are ready to be loaded within 10 seconds from giving the command start. The function of DG and their automatic loading is checked regularly by periodic tests.

Arrangement and layout of sources and networks

The NPP unit, house and reserve transformers are located in front of the unit turbine hall. These locations are separated physically, electrically, as well in terms of the spreading of fire.

The working sources of house consumption (2 house transformers, each with power of 32/16/16 MVA) are supplied from a branch of the main generator. They provide power supply to the 6 kV unit switchboards (BA, BB, BC, BD) that are located in the intermediate machine rooms building (longitudinal and transversal). Reducing transformers 6/0.4 kV and switchboards 0.4 kV to provide power supply to the reactor hall, turbine hall and secondary circuit are installed here, too.

Backup sources of house consumption (2 resreve transformers for each MPU, 40 MVA each) are supplied from the 110 kV network. The reserve sources of MPU I and MPU II can be provided with mutual backup using 6 kV jumpers. Reserve sources provide backup power supply to NPP unit 6 kV switchboards BA, BB, BC, BD.

The 6 kV unit switchboards are used for power supply to 6 kV motors (MCP, feed pumps, coolers, etc.), 6 kV power supply switchboards (BV, BW, BX of systems 1, 2, 3) and transformers 6/0.4 kV for consumers in theturbine and reactor hall. This distribution equipment is located in the longitudinal and transversal intermediate machine rooms.

The 6 kV unit switchboards also supply 6 kV drives and 0.4 kV switchboards installed in external objects. See the following diagram:

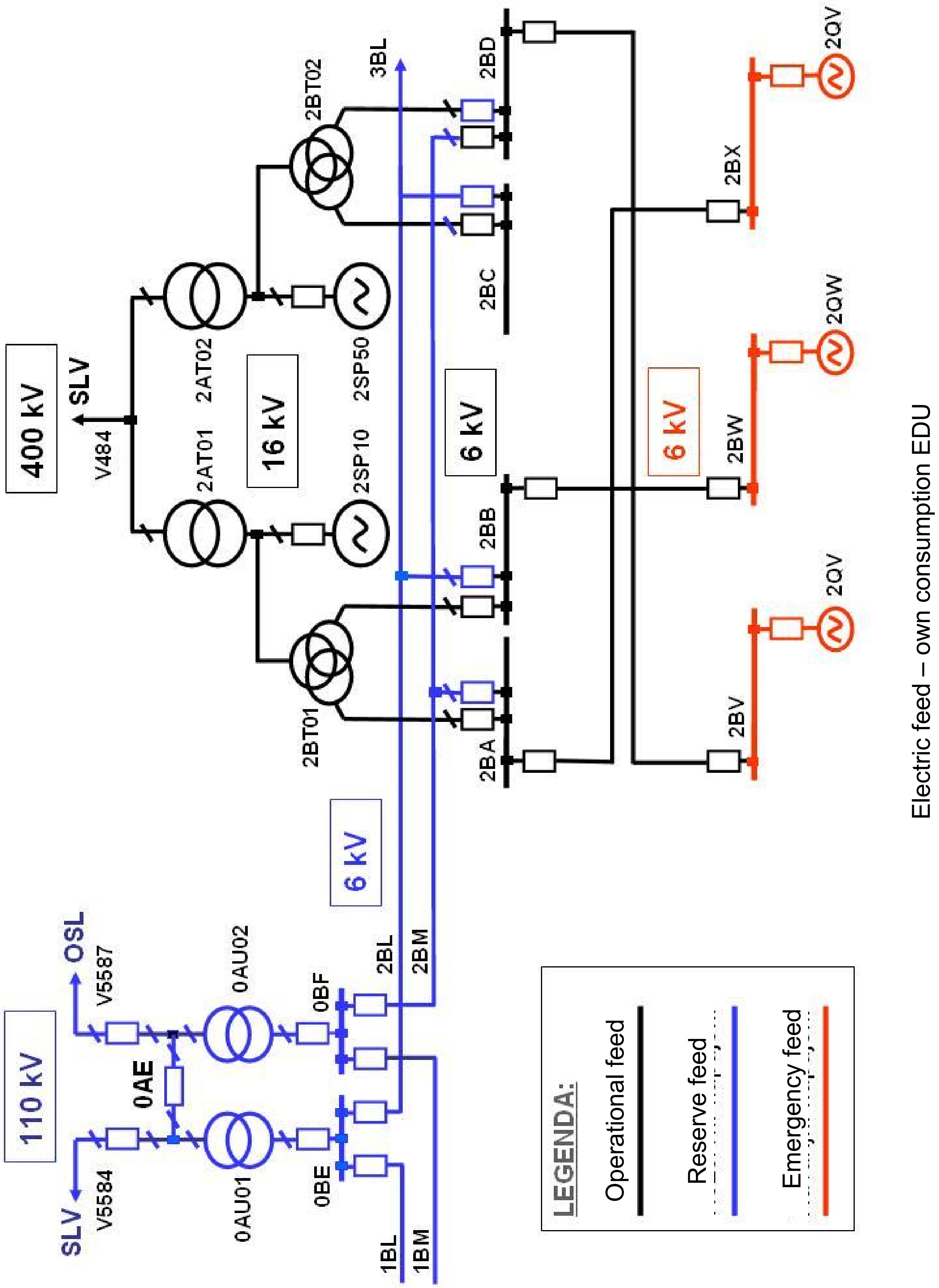


Fig. 14: The basic diagram of NPP Dukovany power supply system HC – 2.RB NPP Dukovany normal design secured power supply systems

Normal design secured power supply systems

*SPSS for power supply of safety systems (*SS*):*

The safety systems (SS) on each NPP Dukovany unit are arranged in 3 divisions of safety systems (3 x 100%). SPSS (marked as 1, 2, 3) is created in compliance with this division and serves as a supporting safety system for power supply of consumers of this division.

Redundancy, separation of redundant sources by structures or distance

These SPSS are independent and separated by building structures against spreading of fire as well as electrically in view of the control system. SPSS 1, 2, 3 are seismic-resistant. Each SPSS has its own emergency power supply sources (DG, batteries) and electrical distribution systems. SPSS 1, 2 also provides power supply to systems SPSS 4.1 and SPSS 4.2 with a lower classification in view of safety (SRS, possibly NSRS) where a high level of reliability and redundancy is not required. However, these systems may not affect negatively fulfillment of safety functions for the SS.

Each SPSS 1, 2, 3 consists of the following main pieces of equipment:

- Emergency DG 6.3 kV, 2.8 MW. Diesel sets (QV, QW, QX) are equipped with their own diesel oil tanks dimensioned for operation at full load for a period of min. 144 hours without refilling (in fact for a longer period because loading is lower). After all, diesel oil can be refilled continuously.
- switchboard 6 kV for secured power supply.
- Switchboards 0.4 kV and reducing transformers 6/0.4 kV.
- Rectifiers, batteries, inverters for power supply of sensitive consumers requiring quality and non-interrupted power supply.

Divisions SS 1, 2, 3 and their SPSS 1, 2, 3 are provided with a backup as a whole (concept 3 x 100%). In view to the principle of independence and mutual separation, any simple fault in any of SPSS 1, 2 ,3 does not cause deterioration of functioning of the remaining two divisions.

DG are emergency sources for consumers permitting interruption of power supply for certain periods (tens of seconds to minutes). DG is started automatically in case of a loss of power supply to the 6 kV switchboard of secured power supply of its SPSS. At the same time, the switchboard is disconnected from the normal power supply system by switching off two in a series of connected sectional switches. Loading of the DG and work of SPSS and consumers are controlled with the highest priority by an automatic device designed for gradual starting (ELS) according to fixed programs. ELS also protects DG against overloading due to continuing development of the accident and following actions of the operator.

SPSS 1, 2, 3 (including DG) consists of seismic-resistant equipment installed in the buildings where improvements of seismic resistance are made.

Objects of DG stations for SPSS 1, 2, 3 are robust reinforced concrete structures located behind the reactor building.

Main electrical equipment of each of SPSS 1, 2, 3 (switchboards 6 kV, 0.4 kV, batteries, etc.) is located in the transversal and longitudinal intermediate machine rooms and thus protected against external risks.

Cable lines of SPSS 1, 2, 3 are mutually independent. This guarantees functional, spatial and fire independence (90 minutes) of these SPSS and respective SS divisions. Cables are segregated in the cable lines according to functional and voltage groups.

All auxiliary systems of the motor and DG (fuel supply to the motor, lubrication oil, inner cooling circuit, filling air, starting air) are autonomous and independent of supplies of external energies during operation of DG. Each DG has its own distributions and sources of own consumption including its own battery. As for the systems that could be influenced by a long-term operation of DG (e.g. clogging of lubrication oil filters), these are redundant sub-systems part of which can be shut down during operation of the DG, to execute necessary maintenance of this component and therefore prevent failure of the DG due to loss of auxiliary systems. The quality of diesel oil is checked regularly.

Thanks to the concept (3 x 100%) of safety systems divisions and SPSS 1, 2, 3 the functional tests of DG and SPSS can be executed even during operation of the NPP unit (test of start of the DG and taking over loads of ELS functions after intentional switching off of the sectional switches and simulation of a real loss of power supply).

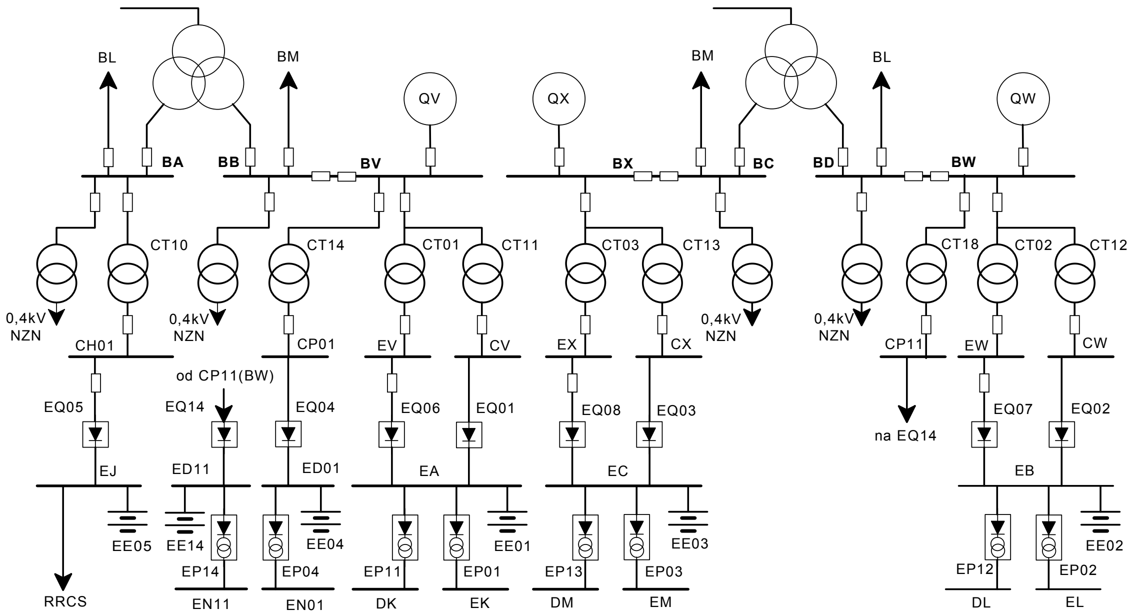


Fig. 15: SPSS systems

SPSS for power supply of systems related to safety (SRS):

A fourth SPSS is installed on each NPP unit to provide power supply to other parts of the systems related to nuclear safety (SRS) and power supply to systems not important in view of nuclear safety (NSRS) that, however, ensure general safety of persons and expensive equipment such as turbine sets. This SPSS is designed as two sub-systems (4.1, 4.2) that provide backup to each other in the principle 100 + 100%. Each of the sub-systems has its own battery, rectifier and inverter. The equipment SPSS 4 is seismic-resistant.

Diesel oil management

Stock of diesel oil in the operating tank of each DG is sufficient for a period of min. 6 hours (4.5 m^3 of fuel at consumption 0.7 m^3/h). Further, each DG is secured by a pair of mutually interconnected tanks with min. stock of fuel 110 m^3. Diesel oil is repumped from the stock tanks to the operating tank automatically when the level in the operating tank drops. Power supply of the fuel transport pumps is provided from the respective DG. The total stock of fuel 114.5 m^3 is sufficient for operation of one DG for a period min. 144 hours (in fact 160 hours), i.e. for 6 to 7 days without any necessity for external refilling of fuel.

Additional fuel for DG can be got by repumping of fuel from other DG units (e.g. that are out of operation) using so-called re-expedition pumps provided that the power supply for their operation is provided (the pumps are supplied from a non-secured power supply source). Then considering long-term operation always only one DG for each NPP unit, putting the re-expedition pumps into operation means stock of fuels without any external delivery of diesel oil to NPP Dukovany.

II.1.1.1.6.4 Alternative secured power supply systems installed permanently on the NPP

Measures and procedures to manage and recovery from complete loss of AC power supply sources of HC (Station Blackout, referred to as SBO) are proposed in the operating rules to maintain the NPP units in a safe condition. The basic method of SBO solution is similar to US NRC RG 1.155. SBO is solved in view of one NPP unit in a situation when the whole power plant is affected by a LOOP.

Definition of the state Station Blackout (SBO) on one of the NPP Dukovany units: SBO is an accident on a NPP Dukovany unit characterized by a loss of all working, backup and emergency sources of AC power supply to the unit – for the unit on power e.g. after disintegration of the electrification system, no regulation any of the two turbine sets to a level of house consumption and no power supply from any of the three diesel sets of the NPP units.

II.1.1.1.7 All diverse sources that can be used

The following power supply sources inside NPP Dukovany are considered for restoration of power supply in the case of SBO:

- utilization of emergency DG of another NPP unit in which at least 2 DG are functional. The operating rules include alternative procedures of operating manipulation, when the arterial reserve 6 kV busbar lines are used for interconnection; the lines run along the whole length of all four NPP Dukovany units.
- a procedure of restoration through power supply lines of the crane track in the reactor hall is prepared for restoration of power supply to the SFSP (TG) cooling pumps.

II.1.1.1.7.1 Other sources and systems for solution of emergency situations

Potential dedicated connections to neighboring units or to nearby power plants

The nuclear power plant Dukovany uses a very robust system of so-called backup power supply, that offers a wide range of options for the power supply of important consumers and restoration of their power supply from neighboring sources after a SBO:

- Basic (verified, within 30 resp. 60 minutes) from the pumping-storage hydro-electric power plant Dalešice, currently using lines 400 kV or 110 kV.
- From the HPS Vranov, using lines 110 kV through the switchyard Znojmo and according to the actual situation either through the 110 kV switchyard Slavětice or through the switchyard Oslavany, that can be used particularly in situations when the switchyard Slavětice is disabled including 110 kV and 400 kV lines between this switchyard and NPP Dukovany due to local natural extreme phenomena (also verified within 60 minutes).
- Power supply from any power supply source connected to switchyards Sokolnice (400, 220 or 110 kV) or Čebín (400 and 110 kV) through the switchyard Oslavany. This is suitable particularly in the case of local disintegrations of 400 kV and 110 kV networks that would disable voltage systems in the switchyard Slavětice.
- Power (including the HPSVranov) through the 110 kV switchyard Slavětice. This is suitable particularly in the case that the 110 kV switchyard Oslavany is disabled, e.g. due to local natural extreme phenomena.

Putting into operation, including a possible repair of some (the least affected) of the above listed lines, can be taken into account within several hours to tens of hours; this is comparable with the time that is necessary for activation (incl. transport) of mobile or other alternative power supply sources

Power supply from mobile sources

Use of mobile sources was not included in the NPP Dukovany design.

II.1.1.1.7.2 Information on each power source

The pumping-storage HPS Dalešice is located approx. 6 km from NPP Dukovany and can be connected through the 400 kV network, or 400 kV and 110 kV through the switchyard Slavětice. In the case that the option is available to reserve some selected 400 kV or 110 kV power supply lines for NPP Dukovany, this could ensure primarily power supply of HC of selected NPP units from the nearby HPS Dalešice (4 x 112.5 MW) or Vranov (3 x 6.3 MW).

This can be realized on the understanding that there will be possibility of communication with the respective external controls and workstations of the hydropower plants. These ways of restoring power supply from external sources are conditioned by serviceability of the 400 kV and 110 kV lines that are not always available.

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

II.1.1.1.7.3 Readiness to take the source in use:

In the case that SBO occurs on a unit that is in hot state, SE NPP Dukovany announces the state EXTREME EMERGENCY that according to the Code of Transmission system of the CR (Act no. 458/2000 coll., Rules of operation of the Transmission system) defines a necessity to provide power supply from an external network to the affected NPP unit within 1 hour. In the case that SBO occurs on an unit that is in a semi-hot state, the state THREAT is announced with a necessity to provide power supply from an external network to the affected NPP unit within 2 hours.

Restoration of power supply to safety-important equipment (provided that the respective external lines and switchyards are serviceable) is provided using the following alternatives:

- Restoration of voltage from the switchyard Slavětice or Oslavany through the 110 kV line.
- Restoration of voltage from the 400 kV system through the 110 kV line.
- Restoration of voltage from the 400 kV system through the NPP unit 400 kV line.
- Restoration of voltage from HPS Dalešice through the 110 kV line.
- Restoration of voltage from HPS Dalešice through the unit 400 kV line (tested in 2004).
- Restoration of voltage from an NPP Dukovany unit through the 110 kV line.
- Restoration of voltage from an NPP Dukovany unit through the unit 400 kV line.
- Restoration of voltage from the hydropower plant Vranov through the unit 110 kV line (tested in 2010).

All the above listed ways of restoring power supply from external sources are conditioned by serviceability of 400 kV and 110 kV lines that are not always available.

II.1.1.1.7.4 Battery sources for DC power supply

Each of the SPSS is equipped with sources and distribution systems that provide uninterrupted power supply to sensitive consumers. Lead-acid batteries 220 V are used as the emergency source. Checks and technical adjustments and coordination of protective and monitoring systems were carried out on all SPSS in view of events on the NPP Forsmark. These checks and adjustment now ensure robust resistance to fault and transition processes in the AC power supply network. In normal operating mode the loads are supplied and batteries recharged using rectifiers from normal power supply sources. In the case of a failure of working and backup sources, power supply of the rectifiers is provided from emergency diesel sets.

The designed rectifiers are able to recharge the batteries within 8 hours.

Description of battery sources

Systems consisting of 2 thyristor rectifiers (220 V, I_n = 800 A, current limitation set to 600 A), lead-acid batteries (220 V, 1500 Ah) and two transistor inverters (220/380 VAC, 160 kVA) are installed on SPSS 1, 2, 3 (classified as SS). These systems provide power supply to the most important control, monitoring and protective systems and fittings of its SS division. Emergency lighting of the rooms is also a significant consumer.

In addition, two sub-systems 48 V (100% + 100%) are installed on each of SPSS 1, 2, 3; these sub-systems provide power supply to the system of protections and control classified as SS (protection of the reactor, ELS automatics, etc.). Each of the sub-systems is equipped with a battery 243 Ah.

The designed discharging time of all batteries on SPSS 1, 2, 3 is 2 hours.

SPSS 4 includes two sub-systems (4.1, 4.2) that are supplied from SPSS 1 and 2. Each of the sub-systems includes a thyristor rectifier (220 V, 800 A), lead-acid battery (220 V, 2000 Ah) and inverter (220/380 VAC, 150 kVA). The 220 VDC are connected by a jumper that allows mutual backup 100% + 100%. These systems are classified as SRS and provide power supply to control system consumers that are classified as SRS or NSRS, and turbine set securing drives. The sub-systems 4.1 and 4.2 are designed as mutual backup (100% + 100%); the consumers are supplied from both sub-systems. In addition, SPSS 4.1 and 4.2 are UPS power supply sources for less important components of the control system and diagnostics.

SPSS 5 includes a battery in the system of reactor control rod drives (220 V, 600 Ah) that stabilizes this system during short-term voltage drops that may occur in the transmission system of house consumption network.

Other safety important battery systems are installed in the diesel set stations. The systems consist of rectifiers and 243 V batteries and their power supply is provided from distribution systems for house consumption of their DG. The systems provide power supply to control systems and protection of DG. Time of discharging by this load exceeds 8 hours. The systems are classified similarly to DG (i.e. DG on SPSS 1, 2, 3 as SS).

Dimensioning in the design of NPP Dukovany, particularly in connection with restoration of I&C, was based on the requirement IAEA NS-G-1.8:2004, i.e. discharging time at least 2 hours.

Tab. 3: Discharging times of batteries acc. to the design on NPP Dukovany

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

x ... 1,2,3 ... acc. to the respective SPSS

Tab. 4: Discharging times of batteries in beyond design basis mode SBO on NPP Dukovany

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

x ... 1,2,3 ... acc. to the respective SPSS

In view of solution of the beyond design basis regime SBO the batteries are not a critical method of connection of the AAC source because:

a) The batteries can be connected to an AAC source that provides power supply to the load and recharging of the batteries using rectifiers,

b) Discharging times of batteries can be extended by a minimal reduction of the load up to a range of 6 to 8 hours, when connection of an AAC source is considered as really practicable and also necessary in view of the technology.

Supplied consumers

Main types of supplied consumers are:

- power supply of the most important control, monitoring and protective systems (RTS, ESFAS, PAMS) and fittings of its SS division (SBSA, PRV, etc.),
- emergency lighting of areas of the given SS division (classified as NSRS) is also an important consumer.

Arrangement and layout of sources and networks

Battery systems SPSS 1, 2, 3 and 4 and the respective rectifiers and inverters are installed in the longitudinal and transversal machine rooms. The systems are seismic-resistant pieces of equipment installed in seismic-resistant rooms.

Alternative options for recharging batteries

Recharging of batteries from AAC sources is considered in the SBO regime.

The current design solution of June 30, 2011 did not consider recharging of batteries in any other way, e.g. from mobile DG. However, this option exists technically.

II.1.1.2 Significant differences between the units

Utilization of design reserves of the units

A project "Utilization of design reserves" of NPP Dukovany units was realized on Unit 3 in 2009 and on Unit 4 in 2010. Within the scope of this design, the nominal thermal power of the reactor was increased from original 1,375 MWt to the value of 1,444 MWt. The project "Utilization of design reserves" also included a reconstruction of high-pressure components of turbines on the respective units. Realization on Unit 1 is planned for the year 2011 and on Unit 2 for the year 2012.

Prevention of complete loss of coolant in case of a LOCA

A modification of the geometry of venting systems line on the inlet to the reactor shaft in the connecting corridor was realized on Unit 3 in 2009 and on Unit 4 in 2010. This modification includes two siphon closures with its lower edge at + 8.1 m, fitted in the upper part with bursting membranes (DN200), with opening pressure 50 kPa on the corridor side, against counterpressure of the fluid in the flooded line. The purpose of the modification is to prevent flowing of coolant to the ventilation centre in the case that the above mentioned level is reached in the reactor shaft. Realization of this modification on Unit 1 is planned in 2011 and on Unit 2 in 2012.

Properties of the reactor pressure vessel (RPV)

A higher value of CDF is present on Unit 1 in the current PSA. This is caused by the fact that RPV of Unit 1 has worse properties in view of brittle fracture. The properties are therefore worse with cold pressurization and worse results of LOCA sequences in which pressurized thermal shocks may occur.

II.1.1.3 Scope and main results of Probabilistic Safety Assessments

First analyses of PSA for NPP Dukovany, Units 1 to 4, were executed in the years 1993 to 1996. The design of analyses of NPP Dukovany covered an analysis of level 1 (PSA Level 1) for production states and a limited set of internal initiating events. These analyses were extended gradually by analyses of other types of risks as well as by non-production states including shutdowns, risks of internal fires and flooding, falls of heavy loads, risks of external events. The analysis of PSA Level 2 was executed in 1998 and later updated, first in 2002 and then in 2006.

Original probabilistic models were updated continuously to reflect the actual state of the design of units after all the gradually realized safety improvements. Updating of the models also included an analysis of fire risks, risk of flooding and updating of models PSA Level 2. The analysis of PSA Level 2 currently includes production operation; PSA Level 2 for non-production operation and shutdowns is currently in the processing stage.

PSA of NPP Dukovany was the subject of the control mission of IAEA IPERS in 1998 (PSA Level 1, internal initiating events, fires, flooding). An independent assessment of the given PSA was also executed (within extent of PSA Level 1 for internal events, for production and non-production operation and PSA Level 2) initiated by the State Office for Nuclear Safety (SÚJB) and executed by an Austrian company ENCONET Consulting in 2005. Control inspections and assessment of the State Office for Nuclear Safety PSA are also executed every year.

Updating of probabilistic PSA models is executed regularly within the scope of a concept Living PSA accepted by the operator, and also as a consequence of requirements of the State Office for Nuclear Safety so that current PSA NPP Dukovany models reflect the current state of the NPP and fulfill basic requirements for their usability for risk informed applications.

An overview of the development of safety expressed in the form of calculated frequency of damage RC (CDF) and frequency of early large releases of radioactive substances for production operation and non-production states and shutdowns is displayed in the following diagram:

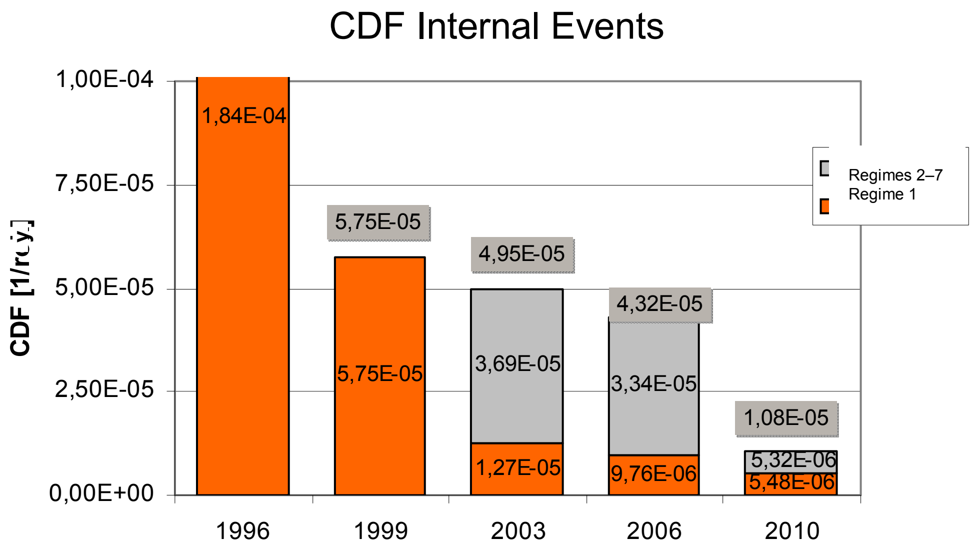


Fig. 16: Development of results of PSA Level 1 (CDF) for production operation of NPP Dukovany (R1) and for non-production regimes/shutdowns (R2-R7), internal initiating evens + fall of aircrafts

CDF is the value of average annual damage to fuel in RC for production and non-production operation of the NPP.

CDF (R1-R7) for the remaining NPP Dukovany units is lower and reaches total values 8.76 ÷ 8.81E-06/yr.

Of the total value of frequency of occurrence of damage to RC (CDF), the contribution of category of early large releases of radioactive substances is approx. 23.5%.

From 2000 to 2003 there were also prepared and since 2005 standardly operated and updated fully functional probabilistic models for monitoring of risks in real time, so-called Safety Monitor. It is utilized for identification and monitoring of risk configurations of all units during shutdowns as well as for monitoring of risk profiles in real time both during operation and during shutdowns of individual units. It is also utilized for assessment of the risk of operation for the purpose of realization of risk informed applications.

Based on current knowledge and when considering external events in the design extent, the following conclusions of PSA Level 1 are applicable:

- A current value of the contribution of a seismic event to the total CDF risk is not available yet; an analysis of the seismic risk is under preparation. The precondition is low contributions to the total risk with regards to the requirement of SSC qualification for a seismic resistance min. 0.1 g and low frequency of occurrence of such intensity of earthquake in the site.
- The contribution of external initiating events caused by human activities to the risk is negligible (the contribution to CDF is about the value 1.0E-7/yr).
- The contribution of emergency sequences leading to SBO due to internal causes is the order of E-7/yr.

II.2 Earthquakes

II.2.1 Design basis

II.2.1.1 Earthquake against which the plant is designed

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

The value of maximum design basis earthquake with the time of occurrence once in 10,000 years determines the level of seismic hazard in the site. For the Dukovany NPP site, this value has been determined, on the basis of the analyses performed, as peak the ground acceleration DBE-2= 0.06 g, which is equivalent to a seismic event with 95% probability of not being exceeded.

The design basis earthquake DBE-1 with the mean time of re-occurrence once in 100 years (determined on the basis of calculations and the map of seismic hazards in the Czech Republic) has been determined for the Dukovany NPP location as the ground acceleration of 0.050 g in the horizontal direction and 0.035 g in the vertical direction, and is equivalent to a seismic event with 90% probability of not being exceeded.

As an assignment for seismic upgrading of Dukovany NPP, in accordance with the international requirements, the value of maximum design earthquake with the mean time of re-occurrence once in 10,000 years has been set at 0.1 g in the horizontal direction and 0.067 g in the vertical direction. Nowadays, the resistance of significant safety equipment and civil structures is being increased to 0.1 g, the value of peak ground acceleration, in all units.

Methodology used to evaluate the design basis earthquake

The territory of the Czech Republic represents an intraplate area and there is no structure of subduction zone type (subduction of the earth plates) here.

Four different approaches have been used to set the earthquake design basis parameters of the maximum design basis earthquake level (DBE-2). Based on comparison of the results of all methodical approaches used, the final values have been set as the most conservative values. The use of a combination of these methodical approaches aims to eliminate the inaccuracies in earthquake catalogues, generalization of epicentre area schemes and to increase the infallibility of the solutions' results.

- Seismostatistical (probability) - worked out in two versions with use of the same earthquake catalogue, but with different composition of epicentre areas.
- Seismogeological (seismotectonic) - based on the assumption that seismic epicentres are connected with active faults.
- Experimental - referred to as "non-zone method", which does not require the definition of source zones and their delimitation, nor determination of seismicity parameters and their seismic potential. It is based on measurement of real attenuation characteristics along the route: the epicentre - the structure under consideration.

The results of the DBE-2 determination by means of the individual methods:

1. Seismostatistical approach – method 1 (DBE-2 = 0.06 g)
2. Seismostatistical approach – method 2 (DBE-2 = 0.09 g)

3. Seismotectonic approach (DBE-2 = 0.06 g)

4. Non-zone method (DBE-2 = 0.05 g)

<u>Seismostatistical approach – method 1</u>

In the course of determining seismic hazard, the presumption is that the tectonic and seismogenerating processes are stable, i.e. it is supposed that the trend of seismic activity which has been observed so far will remain the same in the future. In the calculations, we always assume that an earthquake can occur at any point of each area or at any point of an active fault segment up to the magnitude of the maximum possible earthquake for this area, this fault.

From the point of view of safety, it is thus necessary to consider the least favorable case taking into account, on the one hand, the magnitude of maximum possible earthquakes in individual epicentre areas and, on the other hand, the shortest epicentral distance between the boundary of epicentre areas or an active fault segment and a site.

According to the original document IAEA 50-SG-S1 and in conformity with the recommendation of IAEA NS-G-3.3, seismic hazard of the site has been determined by means of the two following approaches:

- For comparison, the expert estimate based on the map of seismic zoning. However, application of this map for nuclear facilities does not sufficiently meet the requirements, so it is necessary to have a more detailed survey.
- The probability estimate based on a theoretical mathematical model.

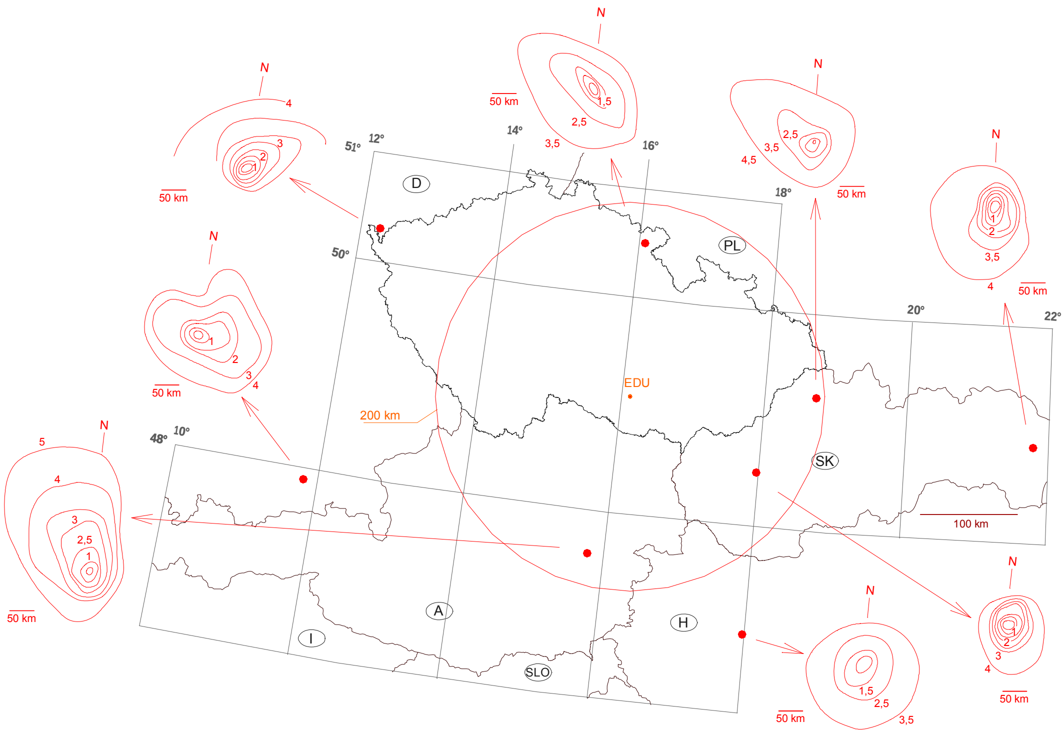


Fig. 17: Shapes of macroseismic fields of source areas in the region of Dukovany NPP

Maximum design basis value data of macroseismic intensity for the observed site, according to the epicentre area for the Dukovany NPP site, are determined by consideration of epicentre areas, values of maximum possible earthquakes that can be produced by the areas in the period of 10,000 years, and by consideration of attenuation curves of macroseismic intensities, which were plotted by the azimuths epicentre area - site, taking into account the shortest distance of an epicentre area from the site (in accordance with the currently used methods in nuclear power, it is the most conservative estimate).

Seismostatistical approach – method 2

The approach of method 2 is based on the calculation of seismic hazard using the probability analysis of seismostatistical and partly seismotectonic input information (seismic hazard probability curve). This method enables us to estimate not only the probability of annual occurrence of various amount of vibratory motions for many years ahead, but also the uncertainty with which these values are encumbered.

The forecast of seismic events is based on the following data:

- Distribution of source zones in the site and in the region.
- Seismicity of source zones and the maximum possible earthquake that can occur in these zones (seismic potential).
- Decrease in the amount of seismic ground movements according to the distance from an epicentre to the site.
- Determination of epicentre areas and their seismicity.

The presumption concerning the validity of historical seismicity parameters as well as an earthquake in the future is grounded in the conception of repeated rough slides on the existing faults. However, we know from experience that new epicentres occur in localities where no historical seismicity has been proven. This presumption is one of the uncertainties in input data.

Source areas of seismic hazard are, for one thing, the epicentre areas of historical earthquakes, and, for another, the lineaments of tectonic faults or their crossing. In Central Europe, the region Dukovany NPP approximately belongs to, 60 epicentre areas have been determined. The seismicity of these areas is expressed by frequency graphs and values of their maximum possible earthquakes (seismic potential).

Other source zones are the faults in the inner part of the Czech Massif, characterized by expertly estimated values of seismicity parameters. Evaluations of the faults together with seismic areas comprise 71 source areas. The seismic hazard of Dukovany NPP has been worked out on the basis of probable earthquake occurrence in these source areas.

Seismogeological (seismotectonic) approach

For the purpose of evaluation of seismic activity of the faults in the site of interest and for the purpose of clear arrangement, the faults are divided into three classes and 6 categories with respect to the amount of magnitude (Mmax) which they can potentially generate. Potential of the faults is evaluated differentially for individual structure units – regional geological units. System of division of the faults into classes and their numeric coding arises from the following chart:

Tab. 5: System of dividing faults into classes and categories on NPP Dukovany

Class	Word designation	Category	M_{max}	I_0 [° MSIS-64]
A	significant seismogenic line	I	6.5	9.5
		II	6.0-6.4	9
B	significant 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 (the maximum design basis values of macroseismic intensity I for the Dukovany NPP site according to a seismoactive fault segment) are determined by consideration of the map of seismoactive faults, the values of maximum possible earthquake intensities that can be produced by seismoactive fault segments in a period of 10,000 years, and by consideration of the attenuation curves of macroseismic intensities that were plotted for the Dukovany NPP site, taking into account the shortest distance of seismoactive fault segments from the location (i.e., in accordance with the current methods, this estimate is the most conservative).

Experimental approach

Experimental determination of seismic hazard is based on use of the so-called “non-zone method”. This method has many advantages; above all, it does not require definition of source zones and their delimitation, or determination of seismicity parameters and their seismic potential.

This method has been applicable only recently, when we have instrument data regarding the acceleration of seismic ground vibrations in the Dukovany NPP site, which are caused by earthquakes in regional distances. The new method does not have to be based only on subjective macroseismic data (historical earthquakes catalogues and maps of isoseismal lines) any more. If we use authentic instrument data, there are none of the uncertainties of the previous methods, which are the consequence of various empirical transfer relations and expert estimates. For example, the relation between the acceleration of seismic vibrations and the local macroseismic intensity is encumbered by uncertainties amounting to two orders of magnitude. Experts consider the new experimental method to be reliable and promising.

By combination of the above-mentioned methods, their inaccuracies have been eliminated. It follows from the results gained by means of various methods that the realistic value of the maximum design basis earthquake of the DBE-2 level for the Dukovany NPP site, which should not be exceeded in the time interval of 10,000 years with the probability ≥ 0.95, is DBE-2 = 0.06 g.

For Dukovany NPP, the DBE-1 design basis earthquake level has been determined on the basis of calculations and the map of seismic hazard in the Czech Republic in correlation with materials for preparation of a building standard on the basis of Eurocode 8. Maps of macroseismic intensities and the PGA_{hor} values show seismic hazard of the Czech Republic with 90% probability that the intensity will not be exceeded in a period of 105 years, for the period of observation of 1,000 years.
On the basis of the above-mentioned maps and calculations, the DBE-1 design basis earthquake level for Dukovany NPP has been determined at the level of 6° intensity of the macroseismic scale MSIS-64 and the acceleration **DBE-1 = 0.05 g.**

II.2.1.1.1.1 Conclusion on the adequacy of the design basis for the earthquake

The level of seismic hazard of the site is determined by the value of maximum design basis earthquake with the time of occurrence once in 10,000 years. The real values of seismic hazard are 0.06 g (with 95% probability of not being exceeded in a period of 10,000 years), or, more precisely, 0.05 g (with 90% probability of not being exceeded in a period of 105 years) for the period of observation of 1,000 years. However, in conformity with the international requirements, the value of maximum design basis earthquake with the mean time of re-occurrence once in 10,000 years has been determined at 0.1 g in the horizontal direction and 0.067 g in the vertical direction.

The value of 0.1 g is the so-called “reference measurement” and it indicates the value of seismic event intensity to which the resistance of the unit should be increased. However, the fact according to seismological analyses and geological examination is that the occurrence of a seismic event the intensity of which is 0.1 g in the Dukovany NPP site is not possible. The above-mentioned corresponds with the fact that if we make approximation of the curves of the seismic hazard of the Dukovany NPP site by production of the curves to values of higher intensities, the frequency of seismic event occurrence with the intensity of 0.1 g can be estimated as lower than 1 x 10^{-8} event per year. It means one occurrence in more than 100 million years.

II.2.1.2 Provisions to protect the plant against the design basis earthquake

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

The deterministic choice of equipment with the necessity of seismic resistance was based on the requirement of meeting prime safety functions: emergency shutdown, ensuring of subcritical state, residual output removal, cooling down, integrity of I.C and II.C, ensuring of sufficient reserve of the I.C coolant and ensuring of the containment leak tightness. To ensure these functions, the necessary systems and equipment, which make up the List of equipment for Dukovany NPP qualification and which have to be seismic resistant, were chosen. The above mentioned equipment and civil structures in which the significant safety equipment is located have been classified as the category “S” based on the requirement of seismic resistance. According to the required function, the equipment of the category “S” is divided into the following subcategories:

Sa (new marking 1a) – Preservation of the full operational capability during and after an earthquake up to and inclusive of maximum design basis earthquake level is required.

Sb (new marking 1b) – Preservation of the mechanical strength and the leak tightness after an earthquake up to and inclusive of maximum design basis earthquake level is required.

Sc (new marking 1c) – Only seismic resistance from the point of view of possible seismic interactions and, above all, preservation of the position stability during and after an earthquake up to and inclusive of maximum design basis earthquake level is required.

In accordance with the Methodology used to evaluate the Dukovany NPP seismic resistance, evaluation of the seismic resistance with use of type tests, calculations or indirect evaluation on the basis of operational experience has been executed for the civil structures and the equipment falling into the category “S”.

Tab. 6: Dukovany NPP civil structures falling into category S with the requirement of seismic resistance:

Name	Seismic category
MPU reactor building	Sb
Longitudinal zone	Sb
Cross zone	Sb
Turbine hall	Sb
DGS	Sb
Venting stack	Sc
SEFWP	Sb
Electro annex building - system 4	Sb
Cooling towers	Sb
CPS	Sb
Electro-channels system pipe channels	Sb

Tab. 7: List of prime machine systems

Name	Seismic category
Continuous primary coolant purification	Sa
ECCS intermediate circuit	Sa
Storage pool cooling	Sa
Hydroaccumulators	Sa
Low pressure emergency core cooling	Sa
High pressure emergency core cooling	Sa
Spray system of HZ	Sa
Essential service water	Sa
Bubbler condenser	Sa
Primary circuit	Sa
MCP	Sa
Volume compensation	Sa
Demiwater 1 MPa	Sb
Dieselgenerator station	Sa
Cooling down system on II.C	Sb
Full-pressure steam system	Sa
Fast-acting valves on the boundary of HZ	Sa
Superemergency feed water supply	Sa
Thermal water treatment	Sb
ESW pumps at CPS	Sa
ESW distribution on II.C	Sa
Ventilation of control centres	Sa
Equipment on the boundary of containment	Sa

Tab. 8: List of prime electro-systems

Name	Seismic category
System 1, 2, 3 of DGS 10, 20, 30	Sa
System 1, 2, 3 of secure power supply of the category II, including regime automatics	Sa
System 4 of secure power supply of the category II	Sa
System 1, 2, 3 of secure power supply of the category I	Sa
System 4 of secure power supply of the category I	Sa
Non-system equipment (distrib. DP10, DP20, DTE3, NDTE1)	Sa

Tab. 9: List of prime I&C systems

Name	Seismic category
Control Room	Sa
Emergency Control Room	Sa
Reactor Trip System	Sa
Engineered Safety Features Actuation System	Sa
n – flow measurement by external chambers	Sa
Reactor Limitation System	Sa
Reactor Control System	Sb
System of E&CR assemblies control	Sa
System for supporting actions	Sa
Emergency Load Sequencer	Sa
Post Accident Monitoring System	Sa
I&C systems in CR	Sa
I&C systems in ECR	Sa
Relay automatics of the primary circuit	Sb
Safety system of the unit	Sa
Relay automatics of the secondary circuit	Sa

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

The designed value of DBE-2 is 0.06 g, whereas the real SSC robustness upgrading is performed to the value of PGA = 0.1 g, which is recommended by the IAEA materials. The decision regarding the Dukovany NPP robustness upgrading to a value higher than the real determined value was made in 1995 in connection with the safety reports of IAEA mission and application of the valid safety instructions IAEA 50-SG-S1 (1991). After full completion of the Dukovany NPP seismic upgrading, the main safety margin will be determined by the difference between the real values of seismic hazard of the site and the design basis assignment of seismic upgrading (PGA of the location is 0.06 g in comparison with PGA_{DBE-}

$_2$ = 0.1 g). It is necessary to regard as another safety margin the fact that these values are upper estimates for 95% probability of not being exceeded. On the basis of the analyses performed, the maximum design basis earthquake level DBE-2 has been set at the value of 0.06 g, whereas the real SSC robustness upgrading is performed at the value of PGA = 0.1 g, which is recommended by the IAEA materials.

The SFSP walls make up part of the seismically resistant containment and even the earthquake equivalent to the PGA = 0.1 g design basis would not lead to a loss of integrity. The assessment has been executed with use of reduced seismic effects by the ductility coefficient, so the value of 0.1 g is, at the same time, the limit value.

The seismic assessment of the civil structures on the containment boundary has proven that, in the case of an earthquake not exceeding 0.1 g, there will not be any failure of leak tightness, even in combination with the load of LOCA accident and with other prescribed loads.

Therefore, it can be stated rightly that after full completion of the design regarding the Dukovany NPP seismic upgrading, the basic safety functions will be preserved up to the level of the ground acceleration of 0.1 g in the horizontal direction and 0.067 g in the vertical direction, which is substantially above the value of real threat to the Dukovany NPP site.

II.2.1.2.3 Main operating provisions to achieve safe shutdown state

In the case of the occurrence of any earthquake in the Dukovany NPP site, the following basic safety functions would not be at risk:

a) Reactivity control

b) Removal of heat from nuclear fuel

c) Confinement ionizing radiation and radionuclides

In the case of equipment failure, the reactor will be shut down by the RTS (HO-1) protection automatically or manually by use of a button. The RTS system is made on the basis of safe failure, i.e., in the case of loss of feed, the E&CR rods will fall down spontaneously into RC and the reactor will thus be shut down.

Cooling down of the unit following a seismic event would be executed in the feed&bleed mode on II.C (feedwater supply of SG through SEFWP + steam removal through SBSA). The water reserve intended for cooling down by feed&bleed on II.C is determined by the quantity of water in the demiwater tanks (they have also been seismically upgraded to 0.1 g). During heat removal through SG into the atmosphere, the quantity of water in these tanks is sufficient for approx. 4 days. The reactors would be put into stabilized semi-hot condition and a decision concerning how to manage their long-time cooling would be made after the inspection of damage to the equipment and buildings (no regulation, procedure or instructions have been worked out so far).

Alternatively, SG feed through the means of LFRU is also being considered, though there might be problems with its accessibility and transport to the spot. If this method of cooling down could not be used in the current condition of seismically non-upgraded MPU building, then an emergency method of cooling down can be used, the so-called feed&bleed on I.C (PSV + ECCS pumps with the ESW support).

Heat removal from the storage pools would be ensured in the same way as it was before the event, i.e., by TG system (100% redundancy).

Integrity of the containment is also not endangered – see the previous chapter.

II.2.1.2.3.1 Protection against indirect effects of the earthquake

The large tanks which might, after their destruction during a seismic event, flood buildings with significant safety equipment (e.g. feed water tanks and ESW storage containers on II.C) have been assessed from the point of view of seismic resistance and they have been, after adjustments performed, anchored sufficiently into a building so that there would be no loss of integrity or undesirable interaction.

A seismic event with the intensity of > 6° MSIS-64 (PGA_{hor} > 0.05 g) would affect seismically non-resistant equipment and structures, which would probably lead to complete loss of power supply for house consumption of all 4 units (the loss of the off-site power supply – 400 kV as well as 110 kV, unsuccessful transition to HC power supply by TG), with following shutdown of all 4 reactors and with transition to natural circulation. Owing to the damage, the nuclear power plant would, very probably, lose the Jihlava Pump station and thus lose raw water supply to the gravity-based water tanks.

In the case of extensive damage to the infrastructure and long-term inaccessibility of the site (collapse of buildings, damage to roads etc.), the relief staff might not be able to get to the site. In this case, the staff who are present at the moment of event occurrence would have to ensure the required activities. Relieving the staff would be solved operatively in cooperation with the state administration bodies (IRS, army etc.).

Data communication facilities inside and outside Dukovany NPP will be endangered in a similar way. As a consequence of the damaged infrastructure in the surroundings of the nuclear power plant, it would be the loss of operability of the technical communication facilities between control centres and persons taking actions, including communication with external control centres and the state administration bodies, which would make actions complicated

In the case of a hypothetical severe earthquake, another problematic part might be the availability of information about Ra situation inside and on the border of the Dukovany NPP premises. All current systems which monitor Ra situation (CRMS, RIS, ISRC, TDS) are not seismic resistant, or their parts are situated in seismically non-resistant service buildings, and so, in the case of an earthquake with the intensity of > 6° MSIS -64 (PGA_{hor} > 0.05 g), it is not proven that they would be in operable condition. In buildings whose resistance has not been upgraded, the radiation control staff might also be endangered. For purposes of radiation measurement, there will still be an alternative way via portable measuring instruments.

Other risks caused by indirect earthquake effects (interaction between components, or more precisely between civil structures) will be known at the end of 2011, when the probability analyses of seismic hazard will be finished.

II.2.1.3 Compliance of the plant with its current licensing basis

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

The earthquake original design basis values of the Dukovany NPP site were changed in 1995 in connection with the safety reports of the IAEA mission. The new seismic assignment was to reassess the significant safety equipment and buildings to a higher level of the reference earthquake PGA = 0.1 g and to perform seismic upgrading of the non-conforming equipment and buildings.

To maintain permanent conformity of the current condition of the equipment with the design basis requirements, a number of activities are carried out regularly with a view to keeping the achieved seismic qualification level of the equipment and buildings.

Licensee's processes to ensure that mobile equipment and supplies are permanently ready for use

In the site, the fire brigade LFRU is permanently present, including mobile equipment for fire fighting, water pumping, salvage of property etc. Nevertheless, they are located in a seismically non-qualified building of the Fire Station. In the case of damage to the Fire Station, the mobile LFRU equipment for firefighting as well as for taking other necessary actions might not be accessible. Considering the fact that occurrence of seismic event is not sudden, but, before its occurrence, partial tremor and symptoms can be observed, the fire equipment can be moved to open spaces in time, where it cannot be damaged by collapsing buildings and equipment. As for this activity, no procedures have been worked out so far.

Access for fire brigades actions (LFRU of Dukovany NPP, or external FB) might be limited owing to collapse of seismically non-upgraded buildings on the inner access road as well as due to fall of debris into the area of the nuclear power plant entrance. Alternatively, the standby entrance into the NPP premises could be used.

However, directly in the site, there is no heavy machinery for removing debris from the main and access roads, which might be buried under debris of seismically non-upgraded buildings. Making the main generating units accessible for mobile equipment would be solved in cooperation with the state administration bodies (IRS, army, etc.).

Similarly, use of some emergency preparedness shelters, including the workplace of the Emergency Staff and the Technical Support Centre, might be endangered because they are located under seismically non-upgraded buildings. In that case, functioning of TSC and ERB would be solved operatively (no detailed instructions are available so far).

Potential deviations from licensing basis and actions to address those deviations

Based on extraordinary checks in terms of seismic resistance, which were carried out in the Dukovany NPP site after the accident in Fukushima, no significant non-conformity of the current state with the design basis requirements has been identified. SSC robustness upgrading is carried out to the value of 0.1 g, which is higher than the maximum design basis earthquake = 0.06 g.

II.2.2 Evaluation of safety margins

II.2.2.1 Range of earthquake leading to severe fuel damage

After seismic upgrading of the equipment and buildings, the border at which loss of basic safety function may occur will rise to the level of an earthquake of the intensity > 7° MSIS-64 (PGA_{hor} > 0.1 g). The seismic event of the intensity > 7° MSIS-64 (PGA_{hor} > 0.1 g) might cause loss of the NPP safety functions. Nevertheless, this is not a value which would be characterized as boundary conditions “cliff edge”, each device has a different value of boundary seismic resistance with a different margin in relation to 0.1 g. With regard to the safety margins, it is expected that some of the safety systems would remain in such a state that would enable the fulfilling of safety functions.

Cooling down during the seismic event > 0.1 g would be still carried out by means of feed&bleed on II.C (SG water feed through SEFWP + steam removal through SBSA), or by emergency cooling down feed&bleed on I.C (PSV + ECCS with ESW support). Resistance of all above mentioned systems and components has already been upgraded to the value of 0.1 g, they have at least 100% redundancy (SEFWP) and are power supplied from secure power supply (from seismically resistant DG).

The value of 0.112 g, which is the boundary CT resistance, can be conservatively called the deterministic limit value of seismic event intensity, the exceeding of which could cause RC damage.

II.2.2.2 Range of earthquake leading to loss of containment integrity

Seismic assessment of civil structures on the boundary of containment has proven that, in the case of an earthquake not exceeding 0.1 g, the leak tightness will not be damaged, even in combination with the loads of LOCA accident and other prescribed loads. Resistance of the containment substantially exceeds the original design requirements.

II.2.2.3 Earthquake exceeding DBE and consequent flooding

The Dukovany NPP site is not endangered by natural or specific floods. The NPP premises are located on a plateau at the altitude of 383.5 - 389.10 meters above sea level, whereas the main civil structures, inside which there is equipment important in terms of nuclear safety, are at the elevation of 389.10 meters above sea level. The roof-like water drain directs to the deep-set watercourses of the Jihlava and the Rokytná Rivers in the direction away from the power plant.

The water reservoir Dalešice lies upstream of the Jihlava River - the dam of the reservoir is at a distance of 4 km from the power plant in the upstream direction. The crest of dam is at the elevation of 384.00 meters above sea level and, when flood water goes through the dam, the maximum water level in the reservoir is at the elevation of 381.50 meters above sea level. A hypothetic break wave from the hydraulic structure Dalešice does not endanger the NPP premises themselves in view of its location at an altitude that exceeds the maximum level of this hydraulic structure with a reserve of approx. 8 m.

The water reservoir Mohelno, whose course in the NPP profile is at the shortest distance of approx. 1 km in the valley under the power plant, has a dam at the distance of approx. 2 km away from the power plant, in the down-stream direction. The crest elevation is 307.15 meters above sea level and the maximum water level is 303.30 meters above sea level, which is approx. 80 m lower than the Dukovany NPP dimension figure of +0,0.

In the case of loss of the Jihlava Pump station due to a break wave following dam damage within a hypothetical earthquake in the Dukovany NPP site, there is sufficient water storage to remove the residual heat.

Other surface watercourses in the surroundings of the NPP are brooks and rivulets with flows lower by at least two orders of magnitude than the Jihlava River. In the Dukovany NPP surroundings, there are no water sources which might cause, in the case of earthquake, flooding of the NPP site.

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

In the Czech Republic, there are no tectonic structures which could result in severe earthquakes. There is 95% probability that the Dukovany NPP site cannot be hit by an earthquake severer than 6° MSIS-64 (PGA_{hor} = 0.06 g). The real SSC robustness is higher, so there is a safety margin for the remaining 5 % uncertainty.

Nevertheless, as early as 1995, a decision was made to perform seismic upgrading of the significant safety equipment and civil structures in the Dukovany NPP area to the value of the peak ground acceleration PGA = 0.1 g (the maximum design basis earthquake, MDE/DBE-2/SSE). This project is still underway. Currently, more than 90% (including all technology) of the significant safety equipment has qualification documentation which complies with requirements and proves seismic resistance, and, as for other equipment, (electro part and I&C systems) implementation of modifications is drawing to an end.

Potential hypothetical consequences of an earthquake are limited to loss of seismically non-resistant SSC which might take part in fulfillment of supporting safety functions. It concerns, above all, possible insufficient mobile equipment capacity, people and loss of communication facilities.

As a consequence of the damaged infrastructure in the surroundings of the nuclear power plant, it would be the loss of operability of the technical communication facilities between control centres and persons taking actions, including communication with external control centres and the state administration bodies, which would make actions complicated

The objective of the proposed measures is further strengthening of the in-depth defence level during an earthquake. Possibilities to improve the in-depth defence are given in the following chart. The chart also includes fields which need preparation of additional analyses since they were not available at the moment of assessing.

Tab. 10: Possibilities to improve the in-depth defence against earthquakes

Opportunity for improvement	Corrective measure	Period (short-term I / medium-term II)	Note
SCC seismic resistance	To complete the project of Dukovany NPP seismic upgrading	II	in progress
SCC seismic resistance	Control and ensuring of non-seismic equipment anchoring	I	
Regulations	To work out earthquake operating regulations	I	
Regulations	EDMG instructions for use of alternative means	II	
Emergency preparedness	To ensure working of emergency response units in case of unavailability of ECC	I	
Analyses	Seismic resistance of LFRU building	I	in progress
Communication	Alternative means of communications after a seismic event	I	
Staff	Analysis regarding threat to shelters on a seismic event	II	
Staff	Ensuring of sufficient amount of staff after a seismic event	I	
Analyses, machinery	Access to buildings, availability of machinery	II	
Analyses	Seismic PSA		PSR report, work in progress

II.3 Flooding

II.3.1 Design basis

II.3.1.1 Flooding against which the plant is designed

Characteristics of the design basis flood (DBF)

The Dukovany NPP site is not endangered by natural or specific floods. The NPP premises are located on a plateau at the altitude of 383.5 - 389.10 meters above sea level, whereas the main civil structures, inside which is the equipment important in terms of nuclear safety, are at the elevation of 389.10 meters above sea level. The roof-like water drain goes to the deep-set watercourses of the Jihlava and the Rokytná Rivers in the direction away from the power plant. Local terrain elevation difference in the area of western towers is resolved by draining possible rural waters through a circumferential drainage ditch into the storm-water drainage system.

The Jihlava River is the nearest watercourse and is also used as a source of service additional water for the NPP. The Jihlava River with the system of water reservoirs Dalešice - Mohelno runs to the north of the power plant, from the north-east towards the south-east - the shortest distance between the power plant and the river is approx. 1 km. The service water draw-off is made from Mohelno reservoir which serves as an equalizing reservoir for the hydraulic structure Dalešice.

The water reservoir Dalešice lies upstream of the Jihlava River - the dam of the reservoir is at the distance of 4 km from the power plant in the upstream direction. The crest of the dam is at the elevation of 384.00 meters above sea level and, when flood water goes through the dam, the maximum water level in the reservoir is at the elevation of 381.50 meters above sea level. The height of the dam is 88 meters.

The water reservoir Mohelno, whose course in the NPP profile is at the shortest distance of approx. 1 km in the valley under the power plant, has a dam at the distance of approx. 2 km from the power plant, in the down-stream direction. The crest elevation is 307.15 meters above sea level and the maximum water level is 303.30 meters above sea level, which is approx. 80 m lower than the Dukovany NPP dimension figure of ±0.0.

Other surface watercourses in the surroundings of the NPP are brooks and rivulets with flows lower by at least two orders of magnitude than the Jihlava River. Small ponds have been built upon them and on their tributaries. The largest of them have water surface with the area of about 0.5 ha and are located in a lower elevation than the power plant premises so they do not endanger these premises.

A hypothetical break wave from the hydraulic structure Dalešice does not endanger the NPP premises themselves in view of its location at an elevation that exceeds the maximum level of this hydraulic structure with a reserve of approx. 8 m. Of all the power plant equipment, only the pump station of raw water on the Jihlava River might be in danger of possible flooding in the case of flood water going through the Jihlava River. The raw water pump station supplies additional industrial water for the Dukovany NPP operation.

In the course of current security supervision on the water reservoirs Dalešice-Mohelno, no significant events and facts which would signal threat to the safety of the water reservoirs have been noted.

The design basis values of maximum daily precipitation amounts according to frequency of re-occurrence are as follows:

Re-occurrence interval [number of years]	100	10 000
Daily precipitation amount [mm]	77	115

The underground water level in the Dukovany NPP premises is several meters below the foundations of the buildings. In this area, refilling and formation of underground water resources is ensured only by atmospheric precipitation infiltration. Natural draining arises in the north and south directions towards the watercourses of the Rivers Jihlava and Rokytná. Local elevation difference of average underground water level of some buildings is resolved by pumping drill holes into the waste-water disposal system. Buildings or rooms containing equipment important from the nuclear safety point of view are not endangered by shallow horizon of underground waters.

Methodology used to evaluate the design basis flood.

According to the Quitt's Methods of Climatic Regionalization, the Dukovany NPP site belongs to a temperate climatic region. In the long-range average, the annual course of precipitation is characterized by the highest precipitation amount in summer months, with its maximum in July (70 mm), and the lowest precipitation amount in winter months, with its minimum in January (21 mm).

The waste-water disposal system is designed as a branched system which ensures gravitational rainwater drainage from the area of approx. 80 ha and, in front of the Dukovany NPP area, is connected to the final rainwater sewage collector.

The catchment areas of the individual sewers were determined by hydrotechnical situation, and an appropriate flow and runoff coefficient, according to the type of building pattern and a kind of land, were assigned to these catchment areas. On the basis of the calculated amount, an appropriate sewer profile was designed. According to hydrotechnical calculations of the storm-water disposal system, the amount of rainwater was set for the periodicity p = 1-minute and 15-minute substitute rain with the intensity of i = 135 l/s.ha and is Q_{rain} = 3.028 l/s, whereas the storm-water disposal system's capacity is 3.810 l/s.

Loading with climatic effects (generally) is based on the statistic elaboration of annual extreme values of relevant meteorological quantities measured for a period of at least 30 years in the Dukovany NPP site and in the meteorological stations in the surrounding region, which have, from the point of view of climatic conditions, the same character as the Dukovany NPP site. The methods of the statistic elaboration are based on the document issued by the International Atomic Energy Agency (IAEA, International Atomic Energy Agency 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 Gumbel's distribution.

In the course of the scheduled Dukovany NPP service life, any changes in the site that would affect the operational safety cannot be, in fact, considered.

The assessment of the effect of torrential rains on the safety of the NPP is based on very conservative assumptions that all storm-water drains (except for downpipes of buildings) will get clogged, and on one-day precipitation with the repeatability after 10,000 years during which water amounting to 115 mm/24 hours will fall. The assessment of the rain effect is based on accumulation of the total precipitation amount in 24 hours (115 mm) on the NPP terrain and on altitude surveying of terrain and roads in the NPP area.

The Annex building of emergency feedwater supply, which is 14 cm higher than the surrounding terrain, followed by DGS 1, which is 17 cm higher than the surrounding terrain, have the lowest location of all building structures with safety equipment, which represents further minimum reserve in case of hypothetical water level increase in the surrounding area by +11.5 cm.

The operational experience with putting into operation and operation of nuclear power plants with VVER reactors can cope with a potential threat to the NPP prime systems and equipment significant for the NPP safety caused by internal floods. The assessment of internal flood effects on the Dukovany NPP operation safety is based on the results of current reports on the area, the results of the Dukovany NPP equipment qualification, experience and approaches to the solution of this problem in nuclear power plants with a similar type of VVER. It follows from the analyses that these events do not prevent safe shutdown and cooling down of reactor units.

Conclusion on the adequacy of protection against external flooding

In relation to the elevation location of the nuclear power plant, watercourses floods do not endanger this site, all civil structures situated within the Dukovany NPP premises are safely protected by the elevation level of these premises.

In the case of flood waters going through the Jihlava River, only the raw water pump station on the Jihlava River (it supplies additional industrial water for the Dukovany NPP operation) might be flooded.

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 the event of theoretically possible short-term precipitation with 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 Dukovany NPP site position excludes the risk of flooding due to natural or specific floods. The Dukovany NPP civil structures are designed to be resistant to flooding on maximum one-day precipitation amount, at which on the surface at the altitude of 389.1 m above sea level the maximum level of water staying is 115 mm, which is the total precipitation amount in 24 hours at 10,000-year maximum. This is only a hypothetical assumption that water will not flow down into the ground space which is laid lower in the surroundings of the Dukovany NPP site. Therefore, it can be stated that even in the case of absolute failure of the storm-water disposal system’s function, a continuous surface of water reaching the level of 115 mm could not occur on the areas subject to review. Nevertheless, Dukovany NPP has been designed with reserve for this level.

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

II.3.1.2 Provisions to protect the plant against the design basis flood

Identification of SSC that are required for achieving safe shutdown state

Of the equipment essential from the point of view of safe shutdown and cooling down of the reactor units and solution of possible abnormal and failure conditions of the Dukovany NPP reactor units, the following civil structures and SSC located therein have been assessed from the point of view of flooding due to extreme precipitation:

1. Turbine halls of MPU I and MPU II (SO 490/1-01, 02) – emergency (auxiliary) feed pumps of the primary circuit cooling down system.

2. Dieselgenerator stations of MPU I and MPU II (SO 530/1-01, 02) – auxiliary circuits, generator excitation.
3. Central pump 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 demiwater pumps and 0.4 MPa demiwater pumps.

It was stated that all above-mentioned civil structures cannot be endangered by floods. The only civil structure containing equipment which relates to the Dukovany NPP operation and which might be affected by flooding resulting from the watercourses is the raw water pump 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 460 m^3/s for Mohelno. 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 the case of failure of the dam of the upper accumulation reservoir it is necessary to expect flooding of the raw water pump station and the loss of raw water supply of the power plant.

The raw water pump 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. The 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.

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 siting of the power station, is the sufficiently dimensioned storm-water disposal system, the above-ground height of entrances, accesses and gates with respect to the surrounding ground space and weathering of the adjacent communications and other outdoor areas adjacent to buildings essential from the point of view of nuclear safety. All the civil structures located in the Dukovany NPP premises are safely protected by means of the altitude level of the premises.

Main operating provisions to prevent flood impact to the plant.

In the case of external floods, performance of none of the following basic safety functions is endangered:

a) Reactivity control.
b) Heat removal from the nuclear fuel.
c) Confinement ionizing radiation and radionuclides.

The control of the reactor RC reactivity and of the spent (nuclear) fuel storage pool (SFSP) is independent of external floods and provides for sufficient subcriticality in the case of possible floods.

In the case of external flood (torrential rain), the function of heat removal is secured through the system of residual heat removal from RC pursuant to procedures for abnormal or accident situations (standard cooling down, 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 of thereof (SFSP) 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 the case of external flood, these systems are not affected.

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

Unlike floods from external causes, internal floods usually have only local character or can be managed easily (by deactivation of pumps) and as for 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 situation in the operating procedure.

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

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 might cause, above all, difficulties concerning transport from the left-bank districts of the city (where most members of the Dukovany NPP staff and employees of the supply companies live) to the right-bank districts (where communications to the power plant are located). It is also necessary to take into account local restriction of passability of the communications in the neighborhood of NPP due to mudslide from the surrounding fields.

In the case of floods and long-term unavailability of the location, the relief staff might not be able to reach the location operatively. In that case, the required activities would have to be secured by the personnel on site during the event. Relieving the staff would be resolved operatively in cooperation with the state administration bodies (IRS, 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 LFRU wireless stations to the Fire Brigade in Třebíč. In this manner, the long-term communication between the persons executing interventions and the control centres might be endangered as well as communication with the off-site centres of the state administration bodies (CCC SÚJB, Regional Emergency Commission, IRS, etc.).

II.3.1.3 Plant compliance with its current licensing basis

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 activities are regularly executed to maintain the required condition of the equipment according to the design. 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 performed as necessary. This concerns inspection of the grates and intercepting traps; possible repairs or replacement are executed depending on their condition.

Licensee's processes to ensure that mobile equipment and supplies are in permanently ready for use

In the Dukovany NPP site the LFRU is available, which has respective equipment at its disposal 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 particularly protected against flooding. However, it is not assumed that in the case of flood in the Dukovany NPP site, use of the mobile fire-suppression equipment would be impossible.

LFRU mobile equipment includes independent means for transport and pumping media, and is also adapted to pump water in case of floods.

Access to certain civil structures might be limited due to floods. With the right function of the storm-water disposal system, there is no risk that the access might be blocked, nevertheless, 1 CPS civil structure is located lower than the 0.0 m level and possibly an increased amount of water could accumulate near the pools under the cooling towers. These situations have not yet been analyzed and no regulations and instructions concerning the ways to proceed are available.

It is likely that it would not be possible to use the emergency preparedness shelters or the workplaces of the Emergency Committee and the Technical Support Centre, which are not protected against flooding. The possible unavailability of the shelters would be resolved operatively. In this case, activity of TSC and ERB does not have to be provided from the emergency shelters.

Potential deviations from licensing basis and actions to address those deviations

During inspection 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. They concerned partial clogging of the storm-water disposal system and are being removed gradually in the work order.

II.3.2 Evaluation of safety margins

II.3.2.1 Estimation of safety margin against flooding

The Dukovany NPP site has never been and even in the future will never be endangered by flooding due to natural floods. Internal floods, in relation to their local character and easy managing, do not endanger nuclear safety (abnormal situation in the operating procedure).

In spite of the fact that the Dukovany NPP site position excludes the risk of flooding due to natural or specific floods, the Dukovany NPP civil structures are designed as resistant to flooding in the case of extreme precipitation, when the storm-water disposal system would be completely taken out of operation. Even the hypothetical example, if all precipitation would not flow off to the ground space located lower in the Dukovany NPP neighborhood and would give a level of water exceeding the maximum design basis value of 115 mm, is not characterized as ultimate conditions, since the foot of most of the civil structures containing significant safety equipment are located even higher above the surrounding ground space (e.g. the annex building of emergency feedwater supply by 140 mm higher than the surrounding ground space, dieselgenerator station buildings by 170 mm higher than the surrounding ground space etc.), which means another minimum reserve of approx. 20% compared to the total precipitation amount in 24 hours at a 10,000-year maximum.

The effects of floods might cause problems only with managing the consequences of floods owing to deteriorated access to the Dukovany NPP site, threat to persons restricted possibility to use the emergency shelters located under the level of communications as a result of extensive floods and damage to the infrastructure in the Dukovany NPP surrounding environment.

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

The objective of the proposed measures is further strengthening of the in-depth defence level during floods. Possibilities to improve the in-depth defence are given in the following chart. The chart also includes fields which need preparation of additional analyses since they were not available at the moment of assessing.

Tab. 11: Possibilities to improve the in-depth defence against flooding on NPP Dukovany

Opportunity for improvement	Corrective measure	Period (short-term I / medium-term II)	Note
Regulations	EDMG instructions for use of alternative means	II	
Analyses	Analysis regarding threat to shelters in the case of floods	II	

II.4 Extreme weather conditions

II.4.1 Design basis

II.4.1.1 Reassessment of weather conditions used as design basis

The original design assumptions are based on the Russian standard PIN AE-5.6, according to which meteorological phenomena with the re-occurrence time of 10,000 years have to be considered. If there are no available sufficient meteorological data relating to the location and if parameters for the 10,000-year re-occurrence time have not been specified, the load is increased by the coefficient 2.5 for extreme wind and by the coefficient 2.0 for extreme snow. Before construction of Dukovany NPP, no particular values of design basis parameters had been specified, so the ČSN standards for common buildings were considered as a basis.
In 2000, the design basis parameters of natural phenomena for the Dukovany NPP site were reassessed so that the load due to natural phenomena was based on the statistical processing of annual extreme values of the relevant meteorological quantities measured over the course of at least 30 years in the Dukovany NPP site and in the meteorological stations in the surrounding region, which have the same character as the Dukovany NPP site as far as climatic conditions are concerned. As for the design basis load due to climatic effects, repeatability of occurrence of the respective phenomenon once every 100 years is considered. As for extreme design basis load due to climatic effects, repeatability of the respective phenomenon once every 10,000 years is considered.

The methods of statistical processing are based on the document issued by the International Atomic Emergency Agency (IAEA) titled “Safety Standards Series” Safety Guide NS-G-3.4: Meteorological events in site evaluation for Nuclear Power Plants, IAEA 2003). The buildings and equipment of the seismic category “S” have to resist the effects of extreme design basis load in such a way so that the function of the systems essential from the point of view of nuclear safety would not be endangered. For other significant safety buildings and equipment, climatic design basis loads have been specified. Specification of parameters for 100-year and 10,000-year load is made according to the IAEA NS-G-3.4 instructions with use of the Gumbel distribution. Based on values of meteorological parameters, design basis load as well as extreme load of buildings and equipment is then specified.

In 2009 - 2010, a calculation check was carried out of the limit values of resistance of the significant safety civil structures to extreme wind conditions; this check will also be carried out for extreme snow conditions. The calculations of the limit values of the civil structures’ load included checks of internal forces of the individual main load-bearing elements of the structure related to the most unfavorable loading combination. Limit resistance of a structure means a load level at which one or more elements of a structural system of a building or a part of such building which is being assessed loses the ability to perform its static function and, as a result of disruption, the function of the significant safety equipment or JE systems might be endangered. In cases where the calculated values were lower than the design basis values or the extreme load, the effect on the equipment located in the civil structures concerned and on the safety functions performed by the respective equipment was reviewed.

For evaluation of the resistance of the civil structures and the equipment to the effects of other natural phenomena, the following extreme climatic effects are considered in the license documentation:

- Wind
- Snow/Ice
- High/Low temperature

Tab. 12: The parameters for 100-year and 10,000-year load (design basis and extreme load) due to effects of natural phenomena

Event (climatic phenomenon) / Parameter	**Time of re-occurrence 100 years**		**Time of re-occurrence 10,000 years**	
	Value	**Load**	**Value**	**Load**
Gust wind / speed	46.2 m/s	0.69 kN/m^2	60.6 m/s	1.26 kN/m^2
Snow / calculated 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 °C 38.5 °C		46.2 °C 46.2 °C	
Minimum temperature / abs. min / year Daily average 5-day average	-30.8 °C -24.0 °C -21.4 °C		-46.7 °C -37.8 °C -35.3 °C	

The value of wind given in the chart is the value of gust (peak) wind speed. Based on this speed, the mean wind speed for the 10-second integration interval is calculated, which is the basis of civil structures' calculations. The mean value for the period of 100 years is 27.4 m/s and 49.1 m/s for the time of re-occurrence of 10,000 years.

The evaluation of the resistance to extreme values of meteorological phenomena has been carried out for significant safety civil structures falling to the seismic category "S".

Tab. 13: Seizmic categories for significant safety civil structures on NPP Dukovany

Marking	Name	Seismic category
800/1-01,02	MPU reactor building	Sb
805/1-01,02	Longitudinal zone	Sb
806/1-01 to 04	Cross zone	Sb
490/1-01,02	Turbine hall	Sb
530/1-01,02	Dieselgenerator station (DGS)	Sb
460/1-01,02	Ventilating stack	Sc
593/1-01,02	SHN annex building	Sb
1-4A2.3	Electro annex building - system 4	Sb
581/1-01,02,05,06	Cooling towers (CT)	Sb
584/1-01,02	Central pump station (CPS)	Sb
	Demiwater tanks 1000 m^3	Sb

High winds

The original assessment of the civil structures regarding wind load was carried out by the deterministic approach in 2000 and by the probability approach in 2008. On the basis of unsatisfactory results, a suspicion arose that the design basis characteristics are specified in an excessively conservative manner and the data from the meteorological station Kuchařovice concerning the wind speed are overvalued.

For the purpose of updating the meteorological data, data from five stations in the Dukovany NPP surroundings for a period of 50 years were obtained. The review carried out in 2010 proved that the original data are not overvalued; on the contrary, based on the review performed, the gust wind is somewhat higher and the specified wind load of the civil structures is substantially higher.

Compared to the basic wind pressure for the respective location according to ČSN 73 0035, which is set at the value of 0.45 kN/m^2 for common civil structures, the basic wind pressure is two times higher as far as the design basis load level is concerned and 3.8 times higher as far as the extreme load level is concerned.

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 snow weight for the respective location according to ČSN 73 0035, which is set at the value of 0.7 kN/m^2 for common civil structures in the locations to the west of Dukovany, the basic snow weight is 1.56 times higher as far as the design basis load level is concerned and 2.78 times higher as far as the extreme load level is concerned.
Compared to the normative load data, these substantially higher snow load data are based on the fact that the snow map in ČSN 73 0035 specifies the 100-year value divided by the coefficient 1.7 and the snow load with the time of re-occurrence of 10,000 years has not been taken into account for common civil structures.

Maximum and minimum temperature

The input sets of measured data for extreme load due to outdoor temperatures effects were 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 calendar years were considered.

Consideration of potential combination of weather conditions

When dealing with combinations of load states, firstly, calculations of individual load states were carried out. The final results of the above-mentioned individual load states were combined to form load combinations. 13 basic load states were determined – these states always considered the operating load, i.e. a permanent load, a random long-term load and partial short-term loads + a load due to one climatic event. The load due to climatic phenomena 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 proposal of load combinations was based on the principle that the operating load is always combined with one extreme load (i.e. with the 10,000-year time of re-occurrence). Combinations with the load due to climatic effects with the 100-year time of re-occurrence were not carried out, since they overlap with the extreme load taken into account. These calculations took the load reliability coefficients 1.0 into account. With respect to the fact that a combination with extreme loads is concerned, in these combinations, the load reliability coefficient 1.0 is taken into account (in a similar way to in combinations with a seismic load). At the same time, a coefficient of a combination according to ČSN 73 0035 is not considered and the combination of load due to extreme effects is not decreased by this coefficient further. The calculations included checks of internal forces of the individual main load-bearing elements of the structure related to the most unfavorable loading combination.

The issue of the concurrent occurrence of two extreme climatic phenomena in the Dukovany NPP site was resolved in 2008, on the basis of the quantitative and qualitative analysis of the climate dynamics in the Czech Republic. The analysis was grounded in the following principle:

Concurrent occurrence of two extreme climatic phenomena is conditioned by mutual dependencies of the two phenomena. Such dependency is the occurrence of a specific meteorological situation in Dukovany NPP enabling occurrence of both or all concurrently considered extreme climatic phenomena. Based on the analysis, only the following two combinations of extreme climatic events were designated as relevant from the point of view of risk:

Tab. 14: Combinations of extreme climatic events on NPP Dukovany

Combination of extreme phenomenon	Upon climatic phenomenon	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 events a) or b), however 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.

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

II.4.2 Evaluation of safety margins

II.4.2.1 Estimation of safety margin against extreme weather conditions

The evaluation focused on a determination of limit conditions under which destruction of the civil structures or failure of the equipment which performs safety functions might occur due to extreme climatic conditions. It was checked whether performance of the following basic safety functions had been ensured:

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

Estimation of safety margin against high winds

The evaluation of the resistance of the civil structures related to the original design basis levels was carried out in 2000. Within the years 2009 - 2010, an independent check of the design basis was carried out, above all, review of the wind load parameters was carried out in 2010. Calculations of the limit wind load were executed for all evaluated civil structures and, as for some civil structures, a limit resistance lower than the resistance which corresponds with the gust wind of the 10,000-year time of re-occurrence was determined.
Due to the effect of extreme wind, a loss of off-site electrical power and reduction of the ability to remove heat to the atmosphere through ESW may occur.

Performance of the reactivity control safety function is ensured even in the case of extreme conditions and no risk related to failure to performance of this safety function has been identified.

With respect to the fact that the ESW safety system does not have 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 event of extreme wind, due to the existing resistance of the CT shell, might result in the reduction of the ability to remove heat trough ESW to the ultimate heat sink for all the 4 Dukovany NPP units at the same time.

In the case of extreme wind it is necessary to take into account a possible failure of the 400 kV and 110 kV grids (the resistance of the lines to a gust wind is approx. 38 m/s), which means transfer to electrical emergency power systems. The loss of the possibility to transfer heat from ESW to the ultimate heat sink would result in the increase of the ESW temperatures and thus in aggravated cooling of DG with a possible risk of failure and a gradual transfer to the SBO condition.

Possible damage to the CPS roof as a consequence of extreme wind might affect the ESW pumps operability as well. With respect to the location of both CPS as well as with respect to the ESW system configuration, however, it is not likely that the Dukovany NPP site would lose all 2 x 12 ESW pumps at the same moment due to extreme wind. A fall of the turbine hall's roof structure due to extreme wind might also mean certain risks with respect to the fact that the turbine hall houses significant safety equipment (the reactor cooling down systems, SG emergency feed water supply, ESW piping, full-pressure steam piping, etc.). In the case of the least favorable development of an accident, this might result in similar risks to those mentioned above (possible loss of ESW, loss of DG cooling).

Possible impact of fall of the reactor hall's roof structure has not been analyzed in detail yet. Possible damage to the fuel deposited in the reactor or in SFSP after the loss of the integrity of the reactor hall's roof due to extreme wind is highly improbable.

The identified risk lies in possible loss of the ability to ensure the removal of heat to the ultimate heat sink from all the 4 reactors and from all the SFSP and in follow-up loss of the ability to monitor the condition of technology after the battery capacity had been used up.

According to the analyses performed, RC might be damaged in approx. 9 hours after the occurrence of SBO if no activity was performed. If the initial state is a shut down unit in the mode 6 (in the calculation, residual power for 10 days after the shutdown was taken into account), the loss of primary coolant will occur in 8-10 hours and the RC will be revealed in 16-20 hours after the loss of flow in II.C on condition that no activity was performed.
To resolve the extreme wind event, the procedure "Destruction of the cooling towers and the 400 kV and 110 kV lines" has been worked out within the scope of AOPs. This procedure has failed to be verified in full according to the analyses performed. As follows from the executed analyses, for full utilization of the strategies mentioned in this procedure, technical modifications would be necessary (construction modifications).

Estimation of safety margin against heavy snowfall and ice

In the course of the executed review of the civil structures load due to extreme snow, an incorrect original calculation regarding the turbine hall's roof load from 2000 was identified. A recalculation of the turbine hall's resistance was carried out in 2010 and the calculated limit resistance of the turbine hall's roof is only 0.95 kN/m^2, i.e. a value lower than the value which corresponds to the level of 100-year snow.
The recalculation of the resistance of the CPS structure and the reactor hall's roof to the extreme snow load has not been finished yet.

The effect of ice on water-management structures has also been analyzed. Within the Dukovany NPP complex, in the water-management structures with free water level, snow cannot occur, due to extremely low temperatures, in such an amount which might endanger their operation, even in the case that some of the units are shut down.
A fall of the turbine hall's roof structure due to extreme snow might also mean certain risks with respect to the fact that the turbine hall has some significant safety equipment (the reactor cooling down systems, SG emergency feed water supply, ESW piping, full-pressure steam piping, etc.). In the case of the least favorable development of an accident, this might

result in similar risks to those mentioned in the previous chapter (possible loss of ESW, loss of DG cooling).

In the case of extreme snow, no immediate effect of extreme weather is expected. Therefore, it is possible by means of simple organizational or technical provisions (continuous removal of the fallen snow, roofs, shelters covering the substantial 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 and only limited capacity of necessary equipment and personnel is available.

Estimation of safety margin against maximum temperature

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

Only if the breakdown of the grid is combined with failure of regulation of all the units to HC, the units would switch to the electrical emergency power systems (DG) and would lose the possibility to top up cool water from the pumping station Jihlava. Even after switching to spraying on CT it would not be possible to maintain the ESW temperature below 33 °C for a long period, which is the value relevant for the technical conditions prescribed for long-term operation of DG (at full output) and the technological condenser. If the ESW temperature is higher than 33 °C, failure of DG does not occur. Continuation of the DG operation would be possible provided that the output is reduced in an adequate manner so that the temperature of lubrication oil (approx. 60°C) and the temperature of the internal circuit coolant (83 °C) are maintained.

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

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

Estimation of safety margin against minimum temperature

In the case of a “long-term” period of low temperature, which has been determined at 35.8 °C for a 5-day period with the time of re-occurrence of 10,000 years for the Dukovany NPP location, all effects and possibilities which have positive effect and are provided by the power plant were taken into account in the course of evaluation. It followed from the result of the evaluation that the systems of heating and protection against freezing are sufficiently dimensioned and operationally secured to be able to cover the needs of heating under conditions of extreme coldness.

The effect of ice in the water-management civil structures has also been analyzed. Within the Dukovany NPP complex, in the water-management civil structures with free water level, occurrence of a quantity of ice that might endanger their operation is eliminated due to extremely low temperatures.

In the case of an extreme low temperature event, performance of the safety functions is ensured.

Conclusion on the adequacy of protection against extreme weather conditions

In the case of extreme wind with the time of re-occurrence of 10,000 years, the safety function of residual heat removal might be endangered. The main cause is that ventilator towers have not been installed in the ESW system and the main cooling towers are not sufficiently resistant to extreme wind. It was also discovered that, in the case of extreme wind occurrence, some significant safety civil structures are not sufficiently resistant; however, detailed effects on the equipment concerned have not yet been analyzed. Possible damage to the fuel deposited in the reactor or in SFSP after the loss of the reactor hall's roof integrity due to extreme wind is highly improbable.
The most significant impact of extreme snow load might be fall of the turbine hall's roof, which might result in a loss of the safety systems located in the turbine hall. The most significant problems might be caused by failure of the ESW system, which might lead to risk to the function of long-term residual heat removal. This holds true on condition that preventive removal of snow off the turbine hall's roof fails. Some partial differences in the actual resistance of selected buildings from the required values of resistance under extreme load are addressed in the project of supplementary seismic qualification of the significant safety equipment in civil structures, which is being completed. Currently, review analyses are underway to re-prove sufficient resistance to the effects of climatic extremes for all civil structures, systems and components which ensure performance of the basic safety functions.

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

On the design basis, the Fire Brigade building (LFRU) is not classified as a significant safety building, therefore, it has not been assessed from the point of view of extreme natural conditions effects (extreme wind, extreme snow, earthquake). Therefore, it is not known whether the LFRU building might be damaged as a consequence of natural conditions. At the present time, analyses regarding the resistance of the LFRU building are being performed.

II.4.2.2 Measures which can be envisaged to increase robustness of the plant against extreme weather conditions

The objective of the proposed measures is further strengthening of the in-depth defence level during extreme natural events. Possibilities to improve the in-depth defence are given in the following chart. The chart also includes fields which need preparation of additional analyses since they were not available at the time of assessment.

Tab. 15: Possibilities to improve the in-depth defence against extreme weather conditions on EDUs

Opportunity for improvement	Corrective measure	Period (short-term I / medium-term II)	Note
Diversion CT	To implement measures for diversion means of the ultimate heat sink (to CT)	II	PSR, in progress
Regulations	To work out operating regulations for extreme events (wind, temperature, snow)	I	
Regulations	EDMG instructions for use of alternative means	II	
Staff	Ensuring of sufficient number of staff after extreme events	I	
Analyses	Resistance of civil structures (LFRU, CPS, MPU etc.) to extreme conditions	I	in progress
Analyses	To work out methods of evaluation of external effects, verification of analyses performed, possible technical measures	II	

II.5 Loss of electrical power and loss of ultimate heat sink

To fully comprehend the following text, it is necessary to be familiar with the content of Chapter II.1 which describes technological systems that ensure the fulfillment of the safety function.

II.5.1 Loss of electricity

NPP Dukovany electrical systems fulfill the requirements of the machine-nuclear part and respect the properties of the electricity network outside NNP, particularly with respect to NPP Dukovany operational safety and the production of electricity.
Ensuring safety in the case of a breakdown in the supply of electricity is ensured by the high level of mutual independence of both working and reserve house consumption resources and the redundancy of the secure supply system (SSPS), which supply key safety systems and components and have their own emergency resources.

The NPP Dukovany house consumption distribution network is supplied from diversified working, reserve and emergency electricity resources.

The **working supply for house consumption** is designed in units. The supply for the house consumption of the unit (HC), via the use of generator switches, can be ensured by two methods:

- By the supply from the 400 kV external network 400 kV (in non-production mode and during downtime)
- By the supply from own TG (during operation of the power unit).

The connection point of Dukovany NPP to the 400 kV external network is Slavětice switchyard, where the output is installed for all four NPP units while ensuring the working supply of house consumption. House consumption for each unit is ensured in a standard manner from the pair of tap-changing transformers connected in the branch from the installation route for each TG output. On the 1st unit, the tap-changing transformer is installed with the same parameters for the common house consumption supply for the whole power plant. These resources are used during normal and abnormal operation and during emergency conditions if still connected to the 400 kV network or the turbo-generator supply.

The working resources are not available if the unit is disconnected due to ongoing preventive maintenance to the 400kV supply system.

NPP Dukovany island operation:

NPP Dukovany units are able to work in the insulated part of the electrification system, i.e. island operation. Island operation is activated by deviation of the network frequency ± 200 mHz measured by FREA16 frequency relay. One part of the TG control system is the specific island regulator (proportional frequency regulator), where the minimum function is to maintain the frequency in the island network. However, after the transfer to the “island”, the features of the nuclear source are related to ensuring nuclear safety. In terms of nuclear safety it is undesirable that during operation in island mode there will be rapid changes in the reactor output. The sufficient “quickly achievable” output is ensured by transferring the passing station into the condenser for regulation of the reserve. The reserve is automatically created from the “Island operation” automated system and is automatically set at up to 20% of each branch opening of the passing station into the condenser. For scenarios related to the high release of TG, the design includes an automatic system for evaluating high TG acceleration and delivering from it the pulse for the accelerator of the TG hydraulic regulation accelerator (the overrun).
A specific procedure was prepared for island operation within the operating regulation for abnormal statuses (P002b). The range of the frequency of the island network where NPP

Dukovany units are able to work in the long term is restricted by setting the 2 levels of frequency relay FREA16 – in the case of exceeding the frequency 47.9 Hz or exceeding the frequency 52.5 Hz (with a 25s delay) there is the automatic disconnection of the unit from the island network and the transfer to house-consumption.

Measurement during the 3.8.2006 event confirmed the high quality of the regulation of the operation in the island operation mode. NPP Dukovany units successfully worked in the actual small surplus island during the disintegration of the UCTE network into three insulated units on 25.7.2006.

The capability of the island operation of NPP Dukovany units is certified as a supporting service for the transmission system operator for the Czech Republic. Detailed information is contained in certification report ICE-OP-28,29,31/2007 for units 1, 2, 4 For unit 3 the certificate ICE-OP-38/2009 is valid. At present, new certification measurement for 1st unit is due to be performed and a new protocol will be issued.

The reserve source of electricity for house-consumption is two 110 kV switchyards - Slavětice and Oslavany - each with two mutually replaceable 110 kV inlets. For the supply there are two spatially and electrically separated 400/110 kV and 220/110 kV transformers and a supply of lines from various directions and nodes in the electrification system. The connection to the external 110 kV network, from which the back-up supply of house consumption (BSHC) is connected, is from the Slavětice and Oslavany 110 kV switchyards; in terms of resources, from the 400/110 kV (220/110 kV) transformer in Slavětice, Sokolnice and Čebín switchyards. The reserve supply for the pair of reactor units (MPU) is ensured by two BSHC transformers connected to 6 kV unit switchyards through reserve busbars. By using switches on the 6 kV side, it is possible to connect the BSHC I. MPU and II. MPU system. This enables the mutual backup of BSHC sources for both MPU. In non-production mode, some of the working or reserve supply may be out of operation for a long time during regular maintenance.

Emergency sources on each unit are three automatically fast starting diesel aggregates (DG, U_n = 6.3 kV, Pn = 2.8 MW, Sn = 3.5 MVA) and the station accumulator batteries SNZ1, 2 and 3, (voltage 220 V, capacity 1500 Ah, and voltage 48 V, capacity 2 x 246 Ah). Uninterrupted energy flows for consumer appliances are ensured by rectifiers and inverters.

There are 3 systems of secure supply indicated as SSPS 1, 2, 3 (3 x 100%) for supplying the safety systems (SS) on each unit. To ensure the necessary rate of redundancy, these SSPS are independent and mutually separated (in terms of construction and fire), both electrically and in terms of the control system.

For the supply to the part of system related to nuclear safety (SRS) and to the systems non related to nuclear safety (NSRS), which ensure the general safety of people and expensive equipment, SSPS 4 is used which is designed as two sub-systems (4.1, 4.2), which are both backed up according to the 100 + 100% principle. The emergency source for each sub-system is a stationary accumulator battery with a capacity of 2000 Ah, 220 V and aggregate for permanent supply. SSPS 4.1 is connected to SSPS 1; SSPS 4.2 is connected to SSPS 2.

II.5.1.1 Loss of off-site power

Loss of electricity may affect one or more NPP Dukovany units. The operation of the unit on output is characteristically higher with the design resistance towards the loss of electricity (in depth additional defense barriers), than during the shutdown for fuel replacement. The worst case scenario in terms of ensuring safety is the loss of electricity to all units at the same time. In terms of the possible configuration of available equipment, the most conservative scenario is the status where one of the units is in shutdown mode.

Design provisions

The loss of the off-site power supply (LOOP) (e.g. during network disintegration accompanied by the loss of 400 kV and 110 kV switchyards) does not cause the automatic transfer to emergency supply sources during the power operation of the unit.

After disconnection of the unit from the 400 kV external network due to external reasons (such as failure in the 400 kV distribution network, which is not related to the nuclear unit failures and the 400 kV lines to the nearest 400 kV switchyard at Slavětice), there will be automatic TG self-regulation at such output that will cover the house consumption of the unit for a long time. The long-term operation of TG for house consumption was tested in practice several times and after the reconstruction of TG within the implementation of the design concerning increased output, it was verified again by testing. The house transformer electrically supplies 4 switchboards for 6 kV house consumption of the unit, from which the main drives for the 1st and the 2nd circuit are supplied as well as the switchyards for the 6 kV secure supply, which electrically drives the safety systems.

If this does not take place (the unit is in downtime, TG do not work or do not regulate or are disconnected), sections of the 6 kV switchboards for the house consumption supply are automatically switched to the 110 kV reserve supply sources (mass automatic backup of the reserve). In this case, the diesel generators (DG) are not started; the accumulator batteries are charged in the standard regime and ensure an uninterrupted supply of DC supply lines.

Only if the above-mentioned automatic systems do not function and there is no automatic switching to the reserve supply, is there a loss of working and reserve unit sources, i.e. the complete loss of the supply of HC (TLHC).

The loss of working and reserve unit sources causes a decrease in voltage on switchboards for a secure 6 kV supply. A TLHC (LOOP) signal is generated, the sectional switches are disconnected and all three DG are started. The sectional switches disconnect the SSPS 6 kV substations for non-secure 6 kV switchboards i.e. from the normal supply network. After starting DG and then connecting to SSPS 6kV switchboards, the key safety drives are gradually activated in accordance with the program for gradual loading of ELS. At the time when switchboards SSPS 6 kV are without voltage, the uninterrupted supply of consumer appliances and distribution line SSPS 1 category are ensured by accumulator batteries. The restoration of charging accumulator batteries is after the connection of DG to the SSPS 6 kV switchboard i.e. maximally within 15 seconds, actually within 10 seconds.

In the case of the loss of the external NPP electricity supply, no essential basic safety functions are put at risk:

a) Control of reactivity.
b) Heat removal from nuclear fuel.
c) Confinement of ionizing radiation and radio nuclides.

NPP Dukovany units can be in the loss of external supply mode maintained in a safe status for a long time or cooled in a cool status or safely kept in the downtime regime (the supply to all required machine systems and I&C systems is ensured) during the start of at least one of three safety DG on each unit.

If during TLHC the unit is in production mode, there is the shutdown of the reactor by effect of the RTS signal and the breakdown of all main circulating pumps (MCP). The residual heat from RC is collected in the natural circulation regime; the collection of steam from SG is through the steam bypass station to atmosphere (SBSA). Water is added into the SG by two emergency supply pumps (EFWP); pumping water from the supply tank (FWT), is added using 1 MPa demiwater pumps from 3 x 1000 m^3 tanks. Alternatively, water can be added into SG from super emergency supply pumps (SEFWP), which pump water from 3 x 1000 m^3 tanks directly into selected SG in the case of failure to complete the SG from EFWP 1, 2.

If during TLHC the unit is in the shutdown status with water cooling, then adding water into the SG is ensured by the cooling pumps in the closed circuit. Alternatively, water can be added into the SG from SEFWP, which pumps water from 3 x 1000 m^3 tanks directly into the selected SG. In the case of an open reactor with a low level of the coolant at the beginning of Mode 6, the activity of both SEFWP is required to prevent the loss of natural circulation.

The storage pool for spent fuel (SFSP) is cooled by two cooling circuits. Each cooling circuit contains a circulating pump and heat exchanger. Heat exchangers are cooled using essential service water. SFSP cooling pumps and ESW pumps are also supplied from DG and automatically start in the program for gradual running of ELS.

All the previously mentioned loads are electrically supplied from DG (3 x 100%) in accordance with the WER concept, where from each SSPS all drives are supplied that are required in the stated mode to fulfill the safety functions. The exception is the mentioned cooling of SFSP, which is only covered by two systems (SSPS 1 and 3).

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

In accordance with the basic concept of the machine-nuclear part (three redundant and independent divisions of safety systems), there are also three redundant and independent assured supply systems (3 x 100 %). Each of these SSPS is a supporting system for the safety systems for the respective division

- SSPS emergency sources for the safety systems are three independent (systems) DG connected to the respective switchboards for secure supply SSPS 6 kV
- Emergency sources for the DC supply are the accumulator batteries which are permanently connected to the respective switchboards. In the case of the loss of working and reserve unit sources and after connection of DG, the accumulator batteries are charged in the standard mode and ensure an uninterrupted supply of DC to the supply distribution lines.

The diesel reserve in the operating tank for each DG is for a minimum period of 6 hours; when pumping the diesel from the storage tanks, the operation of one DG is ensured for a minimum period of 144 hours. Further fuel for DG can be achieved by pumping from the tanks of the other DG. When considering that for long-term operation there is only one DG on each unit, then when commissioning the pumping of reshipped diesel, there is fuel for 18 to 21 days without the external supply of fuel to NPP Dukovany. The steady load on SSPS 1, 2, 3 is less than the nominal DG output (2.8 MW). The only limiting factor for the long-term status of the loss of external sources can be the reserve of diesel. As mentioned above, for each safety DG, there is a reserve of diesel for a minimum of six to seven days of operation without the necessity for an external delivery of fuel.

II.5.1.2 Loss of off-site power and loss of the usual back-up of AC power source

Design provisions

In the case of the loss of the working and reserve supply in parallel to the loss of the emergency supply of the unit (DG) for ensuring the AC supply, there are the following supply sources:

Device used inside NPP Dukovany:

It is proposed to use autonomous sources of the AC electricity supply on NPP Dukovany with the option of simple interconnection through 6 kV reserve busbars, which is described in EOPs or AOPs. If there is SBO only on some NPP Dukovany units and on the other unit there are at least two DG in operation, it is possible that the unit suffered from SBO from the DG of another unit. Therefore there are the following options:

- Restoration of voltage from the NPP Dukovany unit through the 6 kV busbar.
- Restoration of voltage from DG through the 6 kV busbar.

Use of 400 kV or 110 kV lines:

If after events leading to SBO there is possibility to dedicate for NPP Dukovany the selected supply sources of 400 kV or 110 kV, the HC supply would be primarily ensured from selected units from the nearby hydro power plants - Dalešice and Vranov - according to the procedures in AOPs. However, the precondition is the option to communicate with the external workplace (Dalešice pumping hydro power plant, Vranov hydro power plant, Slavětice switchyard, ČEPS, E.ON control centres). The restoration of the supply from the delivery by HPSDalešice (4 x 112.5 MW) or from HPSVranov (3 x 6.3 MW) was repeatedly tested (2004 and 2010) and was proven during tests.

Dalešice hydro power plant was selected after analyzing SBO as the main external AAC source and this function was practically verified by tests. Dalešice power plant (output 4 x 112.5 MW) has the ability to start from a black-out. The test verified the ability to provide a supply within 30 minutes (on a 400kV line) or within 60 minutes (on a 110 kV line).
In the case of SBO on the unit in hot status, SE NPP Dukovany announces the HIGH EMERGENCY status, which according to the Transmission System Code of the Czech Republic defines the necessity to deliver energy from an external network to the affected unit within one hour. If SBO occurs on the unit in a semi-hot status, the THREATENED status is announced with the necessity to deliver energy from an external network to the affected unit within two hours.

The restoration of the electricity supply for key safety equipment (provided that all respective routes and switchyards are in a capable of operation status) is provided by the following variants:

- Restoration of voltage from Oslavany substation through the 110 kV line
- Restoration of voltage from the 400 kV system through the 110 kV line
- Restoration of voltage from the 400 kV system through the 400 kV unit line
- Restoration of voltage from HPS Dalešice through the 110 kV line
- Restoration of voltage from HPS Dalešice through the 400 kV unit line
- Restoration of voltage from the NPP Dukovany unit through the 110 kV line
- Restoration of voltage from the NPP Dukovany unit through the 400 kV line
- Restoration of voltage from Vranov hydro power plant through the 110 kV unit line

All the mentioned methods of restoration of the supply from external sources are conditioned by the operating capability of the required 400 kV and 110 kV line sections.

Battery capacity, duration and options for recharging batteries

Capacity of accumulator batteries SNZ1, 2 and 3 is 1500 Ah. For each SSPS 4.1 and 4.2 the capacity of the accumulator battery is 2000 Ah.
In the case of the loss of working, reserve and emergency unit sources and the accumulator batteries are not charged, an uninterrupted supply of DC supply lines is ensured. Loss of off-site power and loss of the usual back-up AC power sources, and loss of permanently installed diverse back-up AC power sources

SBO is an emergency status for the unit at Dukovany NPP which is characterized by the loss of all working, reserve and emergency sources of AC supply for the unit – on output, e.g. after the disintegration of the electrification system, no regulation of either of the two turbo-generators for house consumption and no supply for any of the three diesel-generators for the unit.

The event connected with the full loss of the supply of the blackout (SBO) type on the unit of NPP Dukovany is highly beyond design and an improbable accident. SBO is resolved in terms of one unit in the situation where the whole power plant is suffering from LOOP. The most serious mode is the origination of SBO on all NPP Dukovany at the same time. This can occur only if all the undermentioned forms of protection at the depth of the electricity supply fail at the same time:

- External working sources – normal supply from the 400 kV Slavětice switchyard
- Internal working sources – non regulation of any of the turbo-generators for house consumption
- External reserve sources – reserve supply from the 110 kV Slavětice switchyard
- External reserve sources – reserve supply from the 110 kV Sokolnice switchyard
- External reserve sources – reserve supply from the 110 kV Čebín switchyard
- Internal reserve resources – supply from HC of the neighboring unit
- All three redundant emergency sources of the AC supply for SSPS 6kV on all four NPP Dukovany units (i.e. in total 12 DG)
- Diverse external source of AC supply from Dalešice hydro power plant through the 110 kV line
- Diverse external source of AC supply from Dalešice hydro power plant through the 400 kV line
- Diverse external source of AC supply from Vranov hydro power plant through the 110 kV line

In the case of SBO there can be (due to the loss of the usual HC NPP Dukovany) the immediate failure of technology in NPP Dukovany external premises which do not have a supply from accumulator batteries, i.e. compressor stations (low pressure, high pressure), pumping stations (CPS 1, 2, pumping station Jihlava), station sources for cooling and other supporting operating units.

The source of electricity in SBO mode is accumulator batteries or local UPS. After exhausting the capacity there is the loss of functions of all measuring and control technology, including the control of the reactivity system (ExCore, InCore), dosimetric systems, computer systems, physical protection system, emergency lighting, electric fire signaling, internal warning system (radio, sirens), telephone and other means of communication.

I&C systems for measuring reactivity (ExCore, InCore) and the system for post accident PAMS1 monitoring are supplied from accumulator batteries SSPS 1, 2, 3.

Battery capacity, duration and options for recharging batteries in this situation

In the case of SBO the discharging of accumulator batteries occurs because there is no source for charging. The discharging time of the safety system batteries is stated by the source of the current load. Capacity of accumulator batteries SSPS 1, 2 and 3 is 1500 Ah. According to the design, the duration of accumulator batteries with the maximum load is a minimum of two 2 hours. When respecting the actual status of the accumulator batteries and the actual load and when considering a decrease in connected consumer appliances, it is reasonable that a duration of several times higher can be expected. It was analytically proven that only the minimum reduction of the load can result in the prolongation of usability up to the range 6 – 8 hours, manuals are processed in EOPs. For further stricter saving of DC sources, all unimportant consumer appliances, including emergency lighting, can be disconnected. A further option for prolonging the supply of consumer appliances SSPS 1,2,3 is to use only one system at the stated time. With this stricter saving it is possible to efficiently prolong the time of usability up to 10-20 hours. A further option is to use the possibility of taking energy from SSPS 4 batteries in the system. For each SSPS 4.1 and 4.2 the capacity of the accumulator batteries is 2000 Ah, the actual duration without performing the principal reduction of the load is about 6 hours. The specific manual and instructions are not yet prepared.

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

On the NPP Dukovany site with the exception of portable LFRU electricity stations there are no available alternative or mobile AC voltage sources that could be used for the resolution of long-term SBO. Nevertheless, there are external resources whose availability and usability for the SBO solution has been verified and tested. Dalešice hydro power plant (output 4 x 112.5 MW) has the ability to perform a black-out start. The test verified the ability to provide a supply within 30 minutes (on a 400 kV line) or within 60 minutes (on a 110 kV line). The ability for black-out start is periodically certified for all four generators of Dalešice hydro power plant and in cooperation with Dukovany NPP this represents the basic method of restoration of the electricity supply after any disintegration of electrification system in the Czech Republic.

Competence of shift staff to make necessary the electrical connections

For performing activities during SBO the EOPs procedure is stated which resolves manipulation for ensuring the heat removal from RC. The manner of the heat removal from SFSP during SBO is not specified in the EOP procedure, although it is resolved as the reference to TSC instructions. At the same time, activities for saving DC sources (accumulator batteries) are described, however, it is desired to further specify and take into consideration the disproportional allocation of individual system loads with specified conditions on the 1st unit.

Activities during long-tem SBO must include the most important savings of DC sources. After disconnection of emergency lighting, it is necessary to replace them with manual lights which are, for work in switchboard and other closed premises, necessary from the beginning of SBO (the emergency lighting is only designed for safe exiting of the workplace).

The capacity of personnel for starting activities during SBO is sufficient; for long-term SBO the use of personnel must be the subject of a special regime (exchange in exposed workplaces, rest, boarding and management of available sources).

The priority activity will be to ensure the heat removal from RC and from SFSP. The cooling of RC and SFSP would first be managed under conducting SE and safety supervisor, in cooperation with the fire brigade and the dosimetric service. After calling TSC, further strategies would be managed on the basis of manual TSC.

The movement of employees during SBO would be restricted by blocked revolving doors and limited by the radiation situation on NPP Dukovany, which, after discharging the accumulator batteries, would only be mapped on the basis of manual measurement.

The good functionality of covers and concentration points is stated by the operational capability of ventilation, capacity of the accumulator batteries and the functionality of technical information and means of communication. The problems are safe places where there is a high concentration of people necessary for actions (including ERB and TSC). These safe places are only supplied from a non secured supply and during SBO the technical means of TSC and ERB are non-functional.

In the case of the breakdown of the supply of the TSPP system from the swithcboard for common 6 kV house consumption, there is automated non-breaking switching to the supply from the central UPS system, TSPP. After discharging of central UPS there is the breakdown of TSPP terminal equipment (alarm system, system for inspection of entrance of per SÚJB closing of TV circuits, issue of identification cards). In addition, the FO NPP Dukovany operation will be ensured according to the internal FO NPP Dukovany documentation.

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

During SBO there is the collection of residual heat from RC in the natural circulation mode by the collection of steam through SBSA which are supplied from accumulator batteries and can be mechanically controlled (from the spot). However, the delivery of water into SG is interrupted and there is a gradual decrease in the level in the SG and a decrease in the heat-exchange area. The ability of the heat removal on II.C is decreased. After de-pressuring of the SG to 0.7 MPa there is automatic (gravitation) discharging of the supply water from FWT (2 x 150 m^3) into SG and the temporary restoration of the heat removal. The heat removal can be ensured in this mode for a period of about 20 hours after the origination of SBO.

If it is not managed to remove the heat from RC by natural circulation through SG, there is the increase of temperature and the pressure in I.C and the consequent opening and collection of the coolant through PRV, which is also supplied from accumulator batteries, into SG boxes (possibly PSV). This causes non-compensated loss of the coolant I.C and the increase of parameters in the SG box.

The increase of temperature in the RC can be prolonged by adding demiwater directly into SG in an alternative manner by using mobile fire water pumps (pressure on the delivery 0.8-1.2 MPa, flow 120-150 t/h). Within the completion of the design, connection points are prepared which enable the fire technology to be connected with the technology. The alternative manner of completion of SG as described in EOPs was practically tested several times and the capacity of this technology was checked for ensuring basic safety functions. In the case of the black-out status on all four NPP Dukovany units at the same time, a certain restriction may be the capacity of the necessary fire technology (emergency plans for the supply of SG for two units with one pump at the same time are not prepared yet). For the completion of demiwater by the optimal flow, the existing reserves of demiwater are available for tanks 3 x 1000 m^3 for each double unit, which, according to the analysis, is sufficient for 72 hours for all 4 units. Together with the use of the reserve of coolant in FWT, adding water into SG of all four units of the NPP, there is a reserve of coolant for about 4 days. For operation during shutdown with a low level of coolant in the reactor, the critical phenomenon is the loss of natural circulation. If adding water into SG does not start within about 4 hours, there is the loss of natural circulation by decreasing the level in the reactor to below the hot leg nozzles. Then the cooling of RC only by the supply of SG would be ineffective because it would not be possible to restore natural circulation through RC. In the case of long-term SBO, to prevent damage of RC in the starting phase of the accident, it is possible to use the coolant from the hydro-accumulator to restore the level in the reactor. A further very effective possibility for keeping the water level in the open reactor is the gravitation filling from the bubbler water trays. The reserve of water for compensation of boiled coolant is about 12 days.

Conclusion on the adequacy of protection against electrical power loss

The sources of NPP Dukovany electricity supply ensure sufficient design robustness and level of protection in the case of external loss of the electricity supply. These are designed with a high level of independence, mutual backup and redundancy (see working and reserve house consumption sources and emergency sources of AC and DC supply, the system of secured power supply – SSPS, which supplies key safety systems and components).

During the operation of the unit there is higher sesign resistance towards the loss of the electricity supply (additional in depth defense barriers) than during the shutdown for the refueling. The worst case scenario in terms of ensuring safety is the loss of electricity to all units at the same time.

In the site are a total of 12 emergency sources of AC supply (DG). In the loss mode for the external supply, NPP Dukovany units can be maintained in the long-term in a safe condition or cooled to a cold status or safely kept in the shutdown mode (the supply of all necessary machine systems and I&C systems), when starting at least one DG on each unit. For each DG there is a reserve of diesel for 6 to 7 days without the necessity to add fuel externally.

In terms of resistance to TLHC in the case of the production mode of the unit, as well as in the shutdown mode, there are a minimum of two independent ways of automatically ensuring the electricity supply of double up to triple redundant systems necessary for the fulfillment of safety functions (through 220 MW alternators after the regulation of TG for house consumption or by means of redundant DG). In the case of low output or disconnected statuses, it is not possible to use TG regulation on HC. Long-term operation under conditions of external supply loss is guaranteed during periods longer than 72 hours.

In the case of full loss of AC supply (SBO), uninterrupted DC voltage (accumulator batteries) is available for the supply of safety systems and systems related to safety emergencies. Without the operation of the respective DG, the accumulator batteries are not charged and the time up to discharging can be between several and tens of hours depending on loading. This value is sufficient for restoration of the supply of HC of the NPP Dukovany unit from the nearby Dalešice or Vranov hydro power plants.

The consequences of long-lasting SBO can be as follows:

If up to the discovery of fuel in RC or in SFSP the AC electricity supply is not ensured or the heat removal by diversified systems or there is no heat removal in an alternative manner from RC or BSP as described in EOPs, in critical cases there could be damage to the fuel in RC or spent fuel stored in the SFSP.

In the case of discharging batteries, due to the loss of functionality of the I&C system, there would be the loss of the ability to control systems and components and the notification of values of important parameters.

After discharging the batteries, there is the loss of lighting and the breakdown of TSPP technology and the prolonging of the period needed for manipulation due to difficult orientation and the movement of personnel.

The loss of operational capability of hardware for communication, warning and notification of personnel will result in the threat to communication between control centres and intervening per SÚJB including communication with external control centres and state administration bodies.

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

The objective of the proposed measures is to strengthen the level of in-depth protection during internal events beyond the framework of the existing design (earthquakes, floods, extreme conditions, results of human activity, etc.) where the result may be SBO:

1. Proposal and implementation of alternative means of the AC supply of the existing equipment for ensuring the cooling and heat removal from RC and SFSP, including the option for connection to the existing electricity distribution system.
2. Proposal and implementation of diversified means for the cooling and heat removal from RC and SFSP, including the option of connection to the existing technology.
3. Proposal and implementation of alternative means for ensuring DC supply and cooling I&C systems for ensuring monitoring of the status and control of selected components.
4. Proposal and implementation of alternative means for the activities and functional communication (internal and external) of personnel.

Opportunities for improving defense in depth are mentioned in the following table. The table also contains the area in which it is necessary to produce additional analyses as these were not available at the time of evaluation.

Tab. 16: Opportunities for improving defense in depth in the case of the loss of electrical power

Opportunity for improvement	Corrective measure	Deadline (short-term I / medium-term II)	Note
Electricity supply I. category	Ensuring additional supply source for systems SPS I. category and selected consumer appliances SPS II. category	II	
Heat removal from RC through II.C	Ensuring additional source for adding water into SG	II	
Heat removal from RC through I.C	Analysis of possibility of alternative adding of water into the reactor by pump and new pipeline	II	
Regulations	Production of the procedure for restoration of SBO supply for all units	I	
Regulations	Production of the procedure for filling of SG of all four units by fire extinguishing technology	I	
Regulations	Filling of the open reactor and SFSP by self-gravitation from channel XL	I	Is in the process of implementation
Regulations	EDMG manuals for the use of alternative means	II	
Analysis	Analysis of the discharging time of accumulator batteries for the unit for releasing the load, filling OIs , changing the connection and operation of emergency lighting	I	Is in the process of implementation
Protection of personnel and communication	Ensuring alternative source of electricity for safe places and telephone exchanges	II	
Movement of personnel	Ensuring alternative source of electricity for the TSPP supply	II	
Personnel	Ensuring sufficient personnel during long-term SBO	I	
Emergency preparedness	Ensuring the functioning of emergency response elements in the case of non-accessibility of ECC	I	

II.5.2 Loss of ultimate heat sink

II.5.2.1 Design provisions

The ultimate heat sink of heat released from the fuel of NPP Dukovany units is the surrounding atmosphere. Unused heat during the production mode of the unit or residual heat after reactor shutdown is transfered into the ultimate heat sink in several ways:

a) Through the secondary circuit by means of the system for condensing and the circulating cooling water – in normal and abnormal operation in the production mode starting and disconnection of TG and in the emergency mode after disconnection of the reactor if working or reserve supply sources are ensured. This method does not ensure the transfer of the reactor into a cold status.

b) Using the system for cooling and delivery of heat into essential technical water (ESW) – at normal and abnormal operation and under emergency conditions, it is possible to transfer the reactor into a cold status (about 50 °C in RC and in SFSP).

c) In the case of direct collection of steam into the atmosphere from SG while filling SG with supply water – under abnormal or emergency operation; it is not possible to transfer the reactor into a cold status (completion of cooling max. to 110 °C).

d) The alternative method of additional cooling in the case of not enabling the natural circulation of the coolant in loops, by the feed&bleed method on the primary circuit (PSV + ECCS) with the heat removal into essential technical water – this strategy is used in all cases of the loss of the heat removal through the secondary circuit. The heat from RC flows directly into containment from which, through the ECCS systems (pumps TH,TQ and cooler TQ) is directed by the ESW system into the atmosphere. In terms of the function and the loss of the ultimate sink, this method is at the same level as the heat removal into the ESW system through the cooling process (similar to b).

The heat from the storage unit pool for spent fuel is via the ESW system.

The heat removal from I.C through secondary parts of the SG is ensured after the shutdown of the reactor while keeping the unit in a hot status or cooling the unit. It is ensured by the flow of supply water into the SG (system for the normal or abnormal or emergency supply) and the collection of steam from the SG into the condensate or into the atmosphere. The heat removal from TG condensers into the circulation cooling water is not evaluated because it does not need to be available (these are not important systems in terms of safety).

For cooling I.C into the cold status, the heat removal from the SFSP spent fuel and for the heat removal from consumer appliances for systems and systems related to nuclear safety, the system for essential technical water is used which transfers the heat into the atmosphere as a controlled heat sink. All three ESW systems are in operation at the same time (redundancy 3 x 100 %).

The ESW system, from the viewpoint of ensuring safety and the transfer of residual heat either from the fuel in the RC or the fuel in SFSP into the ultimate heat sink, has major importance.

Therefore, the evaluation of the loss of the ultimate heat sink is directed to the loss of the ESW system. The heat removal through the secondary circuit can be ensured by a combination of further options in terms of the solution and redundant systems in terms of system configuration. Systems participating in the heat removal are also available in the case of the restriction of the electricity supply for the operation of emergency resources.

The loss of the ultimate heat sink is evaluated from the viewpoint of the entire site. ESW has a two-unit arrangement, with the operation of hot unit pumps from the common route within the division. The loss of ESW causes smaller risks for the heat removal from RC in modes in which the reactor is sealed (all unit modes, with the exception of the heat exchanger), because it is possible to consider the physical principle of natural cooling circulation in the closed I.C with the cooling in SG where it is possible to ensure the heat removal using some of the methods mentioned.

During the replacement of fuel (or in the status where the reactor is not sealed), there is the heat removal through SG in the water mode and the technological condenser into ESW. The loss of the forced flow of ESW causes the fuel located in the open reactor to have similar risks as the fuel in the storage pools.

II.5.2.2 Loss of the primary ultimate heat sink

The loss of the ultimate heat sink cannot threaten the shutdown of the reactor either automatically or manually by use of a button. The creation of a disconnecting concentration can be ensured by the activity of the pumps for the emergency adding of boron (they do not need ESW for operation).

The ultimate heat sink used in NPP Dukovany – atmosphere – cannot be lost. The collection of residual heat from the fuel located in the reactor or the storage pools or SFSP consumer appliances and safety systems into the ultimate heat sink is based on the passive physical principle of the heat transfer from the auxiliary medium into the atmosphere. Therefore, the loss of the ultimate heat sink can be evaluated only as the loss of ability for the transfer of heat, i.e. loss of systems ensuring media flows for the transfer of heat between heat sources and the atmosphere. confinement ionizing radiation and radio nuclides and there is the risk of possible release to the reactor hall and outside the NPP.

The loss of ultimate heat sink cannot threaten the integrity of the containment for modes with the closed reactor. In the case of the open reactor and loss of the heat sink there is no other barier for

The impossibility of adding cooling water into NPP Dukovany and the loss of the heat removal from ESW into the atmosphere

In NPP Dukovany the loss of adding raw water into the volume of circulating cooling water is considered. In this case it is assumed that the heat removal from the ESW system can be ensured into the atmosphere through pouring CT with the use of water reserves in NPP Dukovany. In addition, that it is possible to use the cooler reserve in clarifiers about 5 x 2000 m^3 and the reserves of raw water and in gravitation water tanks with the volume of 4 x 2000 m^3 for the compensation of losses of ESW by evaporation.

For the loss of the CT function (the flow of ESW to CT is not available) in the case of the possibility of adding raw water from pumping station Jihlava, then it is not critical for NPP Dukovany whether the capability of the ESW system to deliver water to consumer appliances remains. By adding raw water it is possible to cool ESW for an unlimited period. The system for adding raw water is not part of the safety system; in the case of TLHC it need not be available.

It results from the analysis of pumping station Jihlava and ESW that in the NPP Dukovany water systems in the case of conservative access (only half CPS I and CPS II is considered, the level in the cooling towers is at the minimum level) there is a reserve for approximately 26 days for production and adding daily water and for about 39 days for the collection of residual heat (operation of ESW pumps) from the disconnected reactor without adding raw water into ESW on NPP Dukovany. The function of heat transfer into the ultimate heat sink is not threatened in terms of safety.

In the case it is impossible to use the flow of ESW in CT for cooling in modes of additional cooling of units, there is accumulation of the heat in available water volumes (ESW system and pools CT), because evaporation from the free CT pools is insufficient for the heat removal from any reactor. During the possibility of adding water from ESW gravity tanks, an acceptable ESW temperature can be kept for longer than 72 hours.

However, in the case of TLHC it will not be possible to consider supplying raw water from pumping station Jihlava. Without adding cool water, there is the heating of ESW. In the case of conservatively considered initial temperature of ESW 29°C there would be heating of ESW to the temperature 33 °C (the design temperature according to the technical conditions for the operation of consumer appliances) after the shutdown of the reactor for about 2 hours. By using the supply of the reserve of water from the clarifiers it is possible to gain a further 3 hours of operation for the only ESW division. An increase of ESW temperature would lead to overheating of TG, which, however, can be successfully compensated by the slight decrease in loading. If this does not take place and no measurements are taken (increase of room ventilation, portable air conditioning system), the overheating of DG could lead to the gradual shutdown of DG.

Loss of ESW system

The ESW system, from the viewpoint of ensuring safety and the transfer of remaining heat either from the fuel in the RC or the fuel in SFSP into the ultimate heat sink, has major importance.

Due to the redundancy of the ESW system 3x 100 % and the further redundancy 2x100 % of each ESW division (4 pumps), the loss of the ability of the transfer of heat from sources is conditioned by the inoperative capability of all ESW pumps (a total of 12 pumps). Due to the special separation of systems and pumps, the independency of the electricity supply and other supporting systems, the existing operating capability of all ESW pumps is extremely improbable. Even during the operation of only one pump in one division of the ESW system, it is possible to ensure the fulfillment of basic safety functions. The only possible reason for the loss of all ESW could be SBO.

In the case of the occurrence of the irreversible release of ESW water which cannot be compensated by switching to another system division, then to slow down the decrease of the level in the ESW well in cases where there is no flow of electricity, it is possible to consider maximum adding of raw water from pumping station Jihlava. Even under the condition that the route from RVD to consumer appliances is not broken, this intervention will only slow down the decrease of the level in wells and within a period of 21 hours there will be the exhaustion of the entire water reserve for the suction of ESW pumps in the respective CPS (at one MPU). The speed of the decrease of the level in the wells depends on the size of the ESW release. In the case of the same event on all units at the same time, this time would be shorter.

Availability of an alternate heat sink

The full loss of ESW means an immediate problem (see also SBO) due to the possibility of the long-term collection of residual heat into the atmosphere through SG after shut down of the units.

The main non-technological means that can be used in the case of the loss of the ultimate heat sink is pumping technology for the company fire brigade. In addition to the possibility of emergency assigning of water into SG through the super-emergency system for adding water (for direct heat removal into the atmosphere), this technology is not adapted for ensuring alternative methods of heat removal from ESW consumer appliances.

In addition to NPP Dukovany technology, there are no other alternative or mobile sources for ensuring the circulation or heat removal from ESW consumer appliances that would be used to improve the response to the loss of heat collection (SBO, release of radioactive substances in the surroundings of the nuclear power plant, etc.).

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

In the case of the tightened reactor, the heat can be collected from the reactor over the long term in the mode for the heat removal through SG, including the option to cool the unit up to a temperature of about 110 °C. Due to the volume of combinations for ensuring this mode, it is diverse from the viewpoint of the solution and also non-redundant from the viewpoint of systems configurations. To ensure the safety function of the heat removal when the reactor is closed, there is no direct risk.

Note: For the heat removal when the reactor is closed, an alternative is to use the heat removal by feed&bleed (i.e. supply of SG by means of LFRU and the collection of steam through SBSA).

When the reactor is opened when the heat removal from RC depends on the operation of ESW (cooling in primary coolant mode, the collection of the heat on the exchanger of the cooling system in the water mode on the secondary circuit), the consequence of the loss of ESW is the increase of the temperature in RC. In this case, it is possible to start filling the pools for the exchange of fuel with the emergency systems for filling I.C with cold water from the ECCS system (in total available up to 1240 m3 of the boric acid solution – according to the status of the technology during the replacement of fuel) and to postpone the increase of the temperature. The heat removal from RC can be kept in this mode for longer than 72 hours. If there is no restoration of the heat removal through ESW, the temperature in the ECCS tanks and in refueling pool may increase up to the saturation limit. The safety status of RC is also kept by a further effective strategy, i.e. keeping the level in the opened reactor by gravitation filling with the coolant from the trays of the bubbler tower. The reserve of coolant for compensation of boiled coolant is about 12 days.

The integrity of the containment cannot be threatened by the loss of ultimate heat sink for modes with the closed reactor. The containment starts to heat but there cannot be the pressure up to the values where its integrity is threatened (designed absolute pressure 250 kPa). The cooling of the containment can be ensured by the activities of ventilation systems in the containment with coolers connected to the cooled water system.

In the case of an accident with the release of coolant from I.C into the containment, its integrity is firstly ensured by spray system as long as they suck from the ECCS tank. After switching the spray pumps into the suction from the floor of the containment, the efficiency of the showering starts to decrease due to the increased temperature on the suction side. In the case of the non-functionality of spray pumps, there is passive showering of the containment by the vacuum-bubbler system available.

Another status occurs in the case of the loss of the ultimate heat sink when opening the reactor (during the shut down for the replacement of fuel) when no other barrier is available for confinement ionizing radiation and radio nuclides. In this case, there is the risk of release to the reactor hall and possibly outside the power plant of Ra substances released from the refueling pool and SFSP coolant while keeping the temperature at the level of the limit of saturation.

II.5.2.3 Loss of the primary ultimate heat sink and the alternate heat sink

External actions foreseen to prevent fuel degradation

The independent means for transporting media is the mobile technology LFRU. Demiwater can be added directly into SG in an alternative manner by using mobile fire water pumps (pressure on delivery of the pump 0.8-1.2 MPa, flow 120-150 t/h). Within the completion of the project, connection points are prepared which enable the fire technology to be connected with the technology. The alternative manner of completion of SG as described in EOPs was practically tested several times and the capacity of this technology was checked for ensuring basic safety functions.

Time available to recover a lost heat sink or to initiate external actions and restore core cooling before fuel damage
See Chapter II.5.2

II.5.2.4 Conclusion on the adequacy of protection against loss of ultimate heat sink

The ultimate heat sink for NPP Dukovany units is the surrounding atmosphere. Unused heat during the production mode of the unit or residual heat after the shutdown of the reactor is collected into the ultimate heat sink – atmosphere – collection in several ways: the transfer of heat between heat sources and the atmosphere is ensured by the ESW system.

On the NPP Dukovany site is a reserve of water sufficient for about 39 days of ESW operation for the collection of residual heat from the disconnected NPP Dukovany reactors without the external adding of water into the ESW system. For one MPU (2 reactors) a total of 12 ESW pumps are available. The loss of all ESW pumps would lead to the loss of the electricity supply on both main units of the stated MPU.
The loss of the ultimate heat sink takes place in the case of the loss of the ESW system to transfer the heat from RC, SFSP and safety system equipment into the surrounding atmosphere. For the heat removal from the RC on the unit in hot or semi-hot status, after the loss of ESW, direct heat removal into the atmosphere through SG is used which is independent to the heat removal by the ESW system. The loss of ESW only relates to the impossibility to cool the unit to the cold status but, in the long term, to keep the unit in the status capable for operation.

In the case of the open reactor when the heat removal from the RC depends on the operation of ESW, the consequence is the loss of ESW and the increase of the temperature of RC. In this case, it is possible to start filling the pools for the exchange of fuel with the emergency systems for filling I.C with cold water from the ECCS system (in total available up to 1240 m3 of the boric acid solution – according to the status of the technology during the replacement of fuel) and to postpone the increase of the temperature. The heat removal from RC can be kept in this mode for longer than 72 hours. If there is no restoration of the heat removal through ESW, the temperature in the ECCS tanks and in refueling pool may increase up to the saturation limit. The safety status of RC is also kept by a further effective strategy, i.e. keeping the level in the opened reactor by gravitation filling with the coolant from the channel of the bubbler tower. The reserve of coolant for compensation of boiled coolant is about 12 days.

The consequences of the unresolved loss of the ability to remove the heat into the ultimate heat sink can be as follows:

Damage to the fuel from RC and the spent fuel located in the SFSP due to the non existence of alternative methods of collection of fuel from RC, SFSP and components cooled by ESW (if it is not possible to add coolant with LFRU technology).

The loss of cooling of emergency sources of the AC supply (DG) in the case of LOOP may cause SBO.

The release of radioactive substances during boiling from the opened reactor during shutdown or from SFSP into the surroundings.

The loss of the capability to control systems and components and the notification of values of important parameters due to the loss of functionality of I&C systems when it is impossible to remove thermal losses from I&C systems equipment.

II.5.2.5 Measures that can be envisaged to increase the robustness of the plant in the case of the loss of the ultimate heat sink

The objective of the proposed measures is to strengthen the level of in-depth protection during internal events beyond the framework of the existing design (earthquakes, floods, extreme conditions, results of human activity, etc.) where the result may be loss of UHS:

1. Proposal and implementation of diversified means for the cooling and heat removal from RC and SFSP, including the option of connection to the existing technology.
2. Description of the use of alternative and diversified means (proposed according to point 1) – i.e. emergency plans (EDMG), with the aim of ensuring the cooling and heat removal from RC and from SFSP.

Opportunities for improving defense in depth are mentioned in the following table. The table also contains the area in which it is necessary to produce additional analyses as these were not available at the time of evaluation.

Tab. 17: Opportunities for improving defense in depth in the case of the loss of UHS

Opportunity for improvement	Corrective measure	Deadline (short-term I / medium-term II)	Note
Heat removal from RC through II.C	Ensuring additional source for adding water into SG	II	
Heat removal from RC through I.C	Analysis of possibility of alternative adding of water into the reactor by pump and new pipeline	II	
Diversified CT	Implementation of measures for diversified means of the ultimate heat sink (to CT)	II	PSR finding
Regulations	Preparation of the procedure for the loss of UHS and ESW systems on all 4 units	I	
Regulations	Completion of the existing regulations with the procedure for filling SG of all four units by fire extinguishing technology	I	
Regulations	The existing regulations prefer the way of filling the open reactor and SFSP with self-gravitation from XL trays	I	Is in the process of implementation
Analyses	The collection of cooling from SFSP by adding coolant and accumulation in TH tanks	I	Is in the process of implementation
Regulations	EDMG manuals for the use of alternative means	II	

II.5.3 Loss of the primary ultimate heat sink, combined with station black out

In the case of an SBO event, ESW pumps are not supplied. Because the means for the transfer of heat into the atmosphere is only ESW, in addition to SBO there is the loss of the forced heat removal from I.C and SFSP into the atmosphere. The SBO event on the NPP Dukovany twin unit automatically means the loss of the ultimate heat sink of the stated twin unit due to the loss of the electricity supply for the ESW pumps.

In the case of the loss of the heat sink on the NPP Dukovany twin unit and, at the same time, the loss of the electricity supply from working and reserve sources, due to the loss of the cooling of DG there is an SBO situation on the stated double unit. The reason is the mutual dependence between DG and ESW – the breakdown of one causes failure in both units.
For the heat removal from the affected units, there is the feed&bleed strategy on the part of II.C. This strategy is based on the possibility of supplying SG on the basis of gravitation from FWT and then by the cooler by means of LFRU and to remove heat from I.C by evaporation of the cooler in the SG and to remove the originated steam through SBSA into the atmosphere. The capability of collecting heat from SFSP is lost fully with the exception of the possibility to add evaporated water by means of LFRU. Due to the fact that no other risks for combination of the loss of ultimate heat sink and SBO were found, the conclusions in the SBO chapter were identified.

Loss of ultimate heat sink due to SBO

The operability of the ESW system depends on the integrity/functionality of CT. The loss of the CT function leads to the decrease in the capability to remove the heat through ESW into the ultimate heat sink. An increase in the temperature of the ESW can lead to the loss of all DG. The problem could only occur in the case of parallel origination of the TLHC status, where there could be up to SBO. The reason is the mutual dependence between DG and ESW – the breakdown of one causes failure in both units.

In the case of the origination of SBO on only one of the twin unit, there need not be the loss of UHS, because the ESW pumps for the side unit will remain in operation. The parallel cooling of both units for which the ESW system is designed is not expected during SBO, so the two remaining ESW can ensure in terms of capacity the heat removal from the unit affected by the SBO event. The possibilities for the use of the ESW flow to the consumer appliances (ECCS coolers, SFSP coolers, technological condensers) for the affected unit, can be exploited due to the breakdown of the pumps (system pumps for normal or emergency additional cooling necessary for keeping the forced flow of medium enabling the heat removal into the ultimate heat sink on the part of I.C or II.C).

Selected consumer appliances for the heat removal from I.C, as well as from SFSP (pumps for cooling SFSP or pumps for additional cooling), can be alternatively supplied from the side unit (the manner of restoration of the supply is described in the valid EOPs procedure). Therefore, a real possibility exists that the collection of the heat from I.C, as well as from SFSP, will be kept and the safety functions will be kept in the long term.

I&C systems in emergency statuses after the shutdown of the reactor are not all in operation, therefore their production of waste heat will be lower. This decreases the demand for cooling the necessary I&C systems. The most important PAMS system has its own cooling which is also connected from the 1st supply category.

An SBO event on both twin units always means the loss of all ESW pumps on the double unit and the loss of medium collecting the residual heat from the coolers on I.C and II.C of the affected units into the atmosphere. There is a strategy available for the heat removal from RC through the filling of SG by gravitation filling from the FWT and LFRU and the collection of steam from SG through SBSA. For SFSP in this mode the heat removal is not ensured in the long term. Without the restoration of heat there would be boiling of the coolant in the SFSP and there could be the opening of the field in the early phase of the accident (for further description, see SAMG). It is possible to use the options for keeping the level of SFSP by means of gravitation fulfillment from the channels of the bubbler tower. The reserve of coolant for adding boiled cooler is about 13 days. The alternative is to use fire technology for adding spent fuel and keeping the temperature of the fuel in SFSP. EOPs mentions this alternative adding of SFSP; the actual procedures for interventions in the area have not been prepared yet.

SBO as a consequence of the loss of ultimate heat sink

The loss of the heat sink will not influence the electricity supply for house consumption in the case that the supply from working or reference sources is ensured. However, in the case of continual loss of the ultimate heat sink, the loss of the external supply and TG for house consumption does not regulate on any of unit on the double unit, there is the starting of emergency sources (DG). On each double unit there are available three DG for odd and three DG for even unit which need the necessary flow of ESW for their operation. For the operation of DG it is necessary to keep the temperature of the lubrication oil (about 60 °C and the temperature of the cooler for the internal circuit (83 °C).

In extraordinary cases, i.e. during emergency supply of the NPP, disintegration of the network, loss of electricity supply of HC of the power plant, etc., where it would be possible to replace DG with another DG or other source, DG could be operated without the protection of the units when only the protection for the loss of oil would remain. Without oil the engine would be damaged and after restoration of delivery of TVS it would not be able to ensure the electricity supply.

After starting DG in 10 seconds within the gradual running program, there is the restoration of the electricity supply of the pair of ESW pumps (of the respective unit and division). If the flow of ESW is not restored, the DG could not be operated in the long term. During the connection of DG to switchboards SPS II. category and loading, there would be gradual heating of the cooling of the DG circuit and the lubrication oil. In the case of the gradual loss of the ultimate heat sink (the heating ESW is due to the impossibility of cooling the heated water through pouring), it is possible to maintain the temperatures by proportionally loading DG. In the case of urgent loss of ESW there is overheating of DG and the loss of operating capability.

II.5.3.1 Time of autonomy of the site before the loss of the normal reactor core cooling condition

See Chapter II.5.2

II.5.3.2 External actions foreseen to prevent fuel degradation

An alternative is to use the fire technology to add the coolant and maintain the temperature in SFSP. EOPs mentions this alternative adding of SFSP; the actual procedures for interventions in the area have not been prepared yet.

II.5.3.3 Measures that which can be envisaged to increase robustness of the plant in the case of the loss of primary ultimate heat sink, combined with station black-out

The objective of the proposed measures is to strengthen protection during initiation procedures beyond the framework of the existing design (earthquakes, floods, external conditions, results of human activity, etc.) where the result can be the loss of the ability to fulfill the safety functions during SBO in the combination with the loss of UHS.

Measures for increasing the robust character of units during the combination of SBO and the loss of UHS are the same as the measures identified in the case of SBO and in the case of the loss of UHS .

II.5.4 Spent fuel storage pools

The residual heat from fuel files saved in SFSP is collected by systems TG11, TG12, and cooled by the 1 and 3 ESW system.

II.5.4.1 Loss of electrical power

In the case of the complete loss of electricity supply (SBO) there is the loss of the forced heat removal from SFSP through ESW into the atmosphere. In the case of SBO on only one unit, it is possible to supply the voltage to system pumps TG11, TG12, from the surrounding unit through the service inlet. The procedure is sufficiently described in EOPs.

In terms of ascertaining a sufficient reserve of a subcritical state, SBO is not a problem. The geometry and the material of the storage grid ensures the sufficient subcritical state and in the case of boiling the fuel or during the filling of SFSP with water without the content of H_3BO_3.

In the case of the loss of the electricity supply, the operational capability of cooling systems SFSP TG11, TG12 is lost. The forced heat removal from SFSP is in the case that SBO is breached and there is a permanent increase of the temperature which is important when filling the upper grid. Without the restoration of the heat removal, there would be an increase in the temperature up to the value of the boiling of the coolant in SFSP. The coolant from SFSP would be gradually evaporated and, if even after this there is none the mentioned methods of filling of boiler coolant, there could be the opening of the fuel in the early phase of the accident.

During the failure of the SBO system, there is no design diversion system available. The heat removal is possible using alternative methods :

- in the case of a leaking reactor, it is possible to ensure delivery of the coolant by discharging the coolant from the hydro accumulators.
- adding SFSP from the above mentioned VBC channels, when it is possible to use the option to maintain the level of SFSP by means of the gravitation filling from the channel of the bubbler tower. The reserve of coolant for adding boiled cooler is about 13 days. EOPs mentions this alternative adding of SFSP; the actual procedures for interventions in the area have not been prepared yet.
- an alternative is to use the fire technology to add the coolant and maintain the temperature in SFSP. In this sense, the pool is easily available for FB technology (through siding corridor). This marginal case of the method of cooling SFSP is based on the filling of SFSP with water supplied into the reactor hall by means of mobile technology with the evaporation of the boiled coolant back into the reactor hall.

Emergency operating regulations, EOPs, mention the above-mentioned alternatives for adding coolant into SFSP, the actual procedures for intervention in the area are not prepared yet.

SFSP are located in the reactor hall outside the containment in the reactor building. All fuel placed into SFSP is sealed in a proven manner, in the case of the occurrence of untightened fuel, this will be located into hermetically sealed containers in the SFSP grid representing a potential barrier for the release of RA substances and passively ensuring the additional cooling of the located fuel assemblies. Therefore, when carrying out alternative activities for the prevention of the damage to the fuel by sufficiently keeping the level in SFSP by adding fuel from SFSP to the reactor hall, this will mean a significant release of RA substances into the area of the reactor hall.

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

The objective of the proposed measures is to strengthen the level of in-depth protection during internal events beyond the framework of the existing design (earthquakes, floods, extreme conditions, results of human activity, etc.) where the result may be SBO:

1. Proposal and implementation of alternative means of the AC supply of the existing equipment for ensuring the cooling and heat removal from RC and SFSP, including the option for connection to the existing electricity distribution system.
2. Proposal and implementation of diversified means for the cooling and heat removal from RC and SFSP, including the option of connection to the existing technology.

Opportunities for improving defense in depth are mentioned in the following table. The table also contains the area in which it is necessary to produce additional analyses as these were not available at the time of evaluation.

Tab. 18: Opportunities for improving defense in depth in the case of the loss of ultimate heat sink combined with the loss of electrical power on NPP Dukovany

Opportunity for improvement	Corrective measure	Deadline (short-term I / medium-term II)	Note
Electricity supply I. category	Ensuring additional supply source for systems SPS I. category and selected consumer appliances SPS II. category	II	
Regulations	Production of the procedure for restoration of SBO supply for all units	I	
Regulations	Filling of the open reactor and SFSP by self-gravitation from XL trays	I	In the process of implementation
Regulations	EDMG manuals for the use of alternative means	II	

The opportunities for improving in-depth protection during SBO, where the consequence can be the loss of the ability to fulfill safety functions, are mentioned above in the table. The table also contains the area in which it is necessary to produce additional analyses as these were not available at the time of evaluation.

II.5.4.2 Loss of ultimate heat sink

In the case of the breakdown of the ESW system there is the loss of the forced heat removal from SFSP through ESW into the atmosphere and it is not possible to remove the heat from SFSP in the standard manner.

From the viewpoint of ensuring the additional reserve of subcriticality, the loss of UHS is not the problem.

In the case of any inability to remove the heat in any combination of pump – exchanger – ESW, it is possible to ensure the heat removal by discharging the coolant from SFSP into TH tanks of the ECCS system using TG10D02 pump with the heating of water in these tanks. There is no undesired mixing of media because TH tanks low-pressure are designated for fulfillment in modes of fuel replacement (the solution from these tanks serves to increase the level in SFSP in the mode for replacement of the fuel). This procedure would be applied up to the increase of the temperature in all ECCS tanks to 60 °C. Using pumps TM13(14)D01 water is pumped into SFSP. In this manner it is possible to prolong the time it takes to reach boiling point in the pool. In the case of insufficient cooling of SFSP, a procedure is prepared which is part of the EOPs.

A further method of heat removal from the pool is considered to be the adding of SFSP from VBC channels that can be ensured by using the pump for filling the VBC XL10D01 channels.

Another way of filling SFSP is the possibility to use pumps TM13(14)D01TM, which serve for cleaning the SFSP coolant, and using these pumps to deliver the coolant from tanks of low-pressure ECCS (TH) into SFSP. TM pumps are supplied from 2nd category system sources.

When the reactor is open during replacement of the fuel, there is the possibility to supply the cooling by any high-pressure or low-pressure pump from the ECCS system directly into the reactor connected to SFSP and from there to I.C, or it is possible to ensure the delivery of the coolant by discharging from the hydraulic accumulators.

When using the coolant from all ECCS tanks and bubbler water trays, the reserve of coolant is sufficient for adding losses from boiling of the coolant in the SFSP for more than 8 days even in the case of the arrangement of the fuel in two grids, one above the other.

If any of the mentioned methods cannot be used, then cooling of SFSP is possible by alternative means, as in the case of SBO:

- adding SFSP from the higher located VBC channels by gravitation
- when the reactor is open it is possible to supply coolant by discharging from hydraulic accumulators
- for filling SFSP with water by means of LFRU

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

The objective of the proposed measures is to strengthen the level of in-depth protection during internal events beyond the framework of the existing design (earthquakes, floods, extreme conditions, results of human activity, etc.) where the result may be UHS:

1. Proposal and implementation of diversified means for the cooling and heat removal from RC and SFSP, including the option of connection to the existing technology.
2. Description of the use of alternative and diversified means (proposed according to point 1) – i.e. emergency plans (EDMG), with the aim of ensuring the cooling and heat removal from RC and from SFSP.

Opportunities for improving defense in depth are mentioned in the following table. The table also contains the area in which it is necessary to produce additional analyses as these were not available at the time of evaluation.

Tab. 19: Opportunities for improving defense in depth against loss of UHS on NPP Dukovany

Opportunity for improvement	Corrective measure	Deadline (short-term I / medium-term II)	Note
Diversified CT	Implementation of measures for diversified means of the ultimate heat sink (to CT)	II	PSR finding
Regulations	Preparation of the procedure for the loss of UHS and ESW systems on all 4 units	I	
Regulations	The existing regulations prefer the manner of filling the open reactor and SFSP with self-gravitation from XL trays	I	In the process of implementation
Analyses	The collection of cooling from SFSP by adding coolant and accumulation in TH tanks	I	In the process of implementation
Regulations	EDMG manuals for the use of alternative means	II	

II.5.4.3 Loss of ultimate heat sink, combined with station black-out

This case fully corresponds to the status described in II.5.2.1, because, at the same time as SBO there is a loss in the ESW (UHS) system.

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

This case fully corresponds to the status described in II.5.2.1, because, at the same time a SBO there is a loss in the ESW (UHS) system.

II.6 Severe accident management

II.6.1 Organization and arrangement of the licensee to manage accidents

II.6.1.1 Organization of the licensee to manage accidents

The functioning severe accident management system is ensured by a combination of personnel, administrative and technical actions. In the personnel area, this concerns the creation of the emergency response team and ensuring activities assigned to individual positions; in the administrative areas this concerns the preparation and implementation of the respective procedures, manuals and instructions; in the technical areas this concerns ensuring the functionality of the required scope of hardware for implementation of the strategy and for creating the structure of emergency support centres from which the personnel performs the management and execution of these actions.

Staffing and shift management in normal operation

Operation of all NPP Dukovany units is carried out by the shift personnel. The number of personnel in each shift and their qualification allows the managing of all operating statuses of units under normal, abnormal and emergency operating conditions. Shifts are regularly altered according to the shift time schedule so the operating personnel have sufficient time to rest and maintain the required level of qualification (training, verification of knowledge, health and psychical capability, …).
The shift personnel carry out all activities according to operating documents (procedures, instructions, programs, …), covering normal and abnormal operation and emergency conditions (all design basis and partial beyond design basis events up to damage to the fuel). In all these operating statuses, the shift personnel manage and execute the activities with the support of the other technical personnel in the nuclear power plant. In the case of the origination of emergency conditions with damage to the fuel, the responsibility for managing the activities is transferred to TSC and ERB personnel and the shift personnel continue to execute activities according to the requirements of TSC and ERB.

The operational management of the whole NPP is ensured by the Shift Engineer (SE); they are deputized in the shift by a safety engineer.

The management of each unit of the NPP in the case of the origination of an extraordinary event is ensured by the following work positions:

- Reactor unit supervisor (RUS)
- Reactor operator
- Turbine operator.

The main workplace for these personnel is the respective unit control room. In the case of non occupation or the loss of control of unit technology, they perform their activities from the emergency control room.

On the affected NPP Dukovany unit, during the announcement of the extraordinary event (EE), the CR personnel is completed by the safety supervisor, who takes over the responsibility for managing the technology and becomes the contact person between the head of emergency board and the unit control room for the affected unit.

Plans for strengthening the site organization for accident management

The accident management strategy is derived from the logical development of any event in the NPP. In the case of the origination of an extraordinary event, the respective actions are prepared for the requirements of the management and for intervention or intervention instructions for employees or other persons in the selected work positions classified into the Emergency response organization (ERO).
Intervention during the origination of an extraordinary event in the NPP is always ensured in the first phase of the development of an extraordinary event by personnel from the 24 hour shift operation (IOER – internal organization of emergency response), under the management of the SE.

In cases where the event is, by its scope, outside the framework of the personnel of the 24 hour shift operation, IOER is completed by employees ready for operation within the organization of an emergency response (SOER – standby emergency response organizations). In this case, the emergency support centres are activated: Emergency board, technical support centre, external emergency support centre, emergency information centre and the logistic support centre. The responsibility for the management of actions after activation of ERB is taken from SE by the Commander of ERB.

During the origination of EE, the SE ensures the immediate notification of management of NPP Dukovany and ČEZ, immediately reports the event to SÚJB, Regional Government, Regional Directorate of FB, those municipalities with an expanded scope of authority, to Technological Dispatcher of ČEZ and to the meteostation. For handing over information, the form “Primary notification, or consequent reports on the origination of an extraordinary situation” is used. Electronic mail is used for sending forms or faxes. If impossible to establish a direct connection with SÚJB then an alternative method is used.

For managing extraordinary situations, an emergency response team is created where the internal part (IOER) consists of shift personnel and the on-call section (SOER) consisting of technical experts of the NPP who should be ready for action (in 4 crews). The SOER members are on-call so that within 20 minutes during working hours and within 1 hour during off-working hours from the announcement of an extraordinary emergency event, the respective experts are in attendance at NPP Dukovany to ECC. Means for activation of SOER personnel are backup.

To enable the planning of protection of inhabitants in the surroundings of NPP in the case of a radiation accident and for the requirements for preparation of an external emergency plan, on the basis of the decision of the SÚJB a zone for emergency planning of the NPP for NPP Dukovany is declared with a radius of 20 km. For ensuring measures for the preparation and actual evacuation of inhabitants, this decision states the internal part of EPZ stated by a ring with the radius of 10 km including the municipalities on its border.

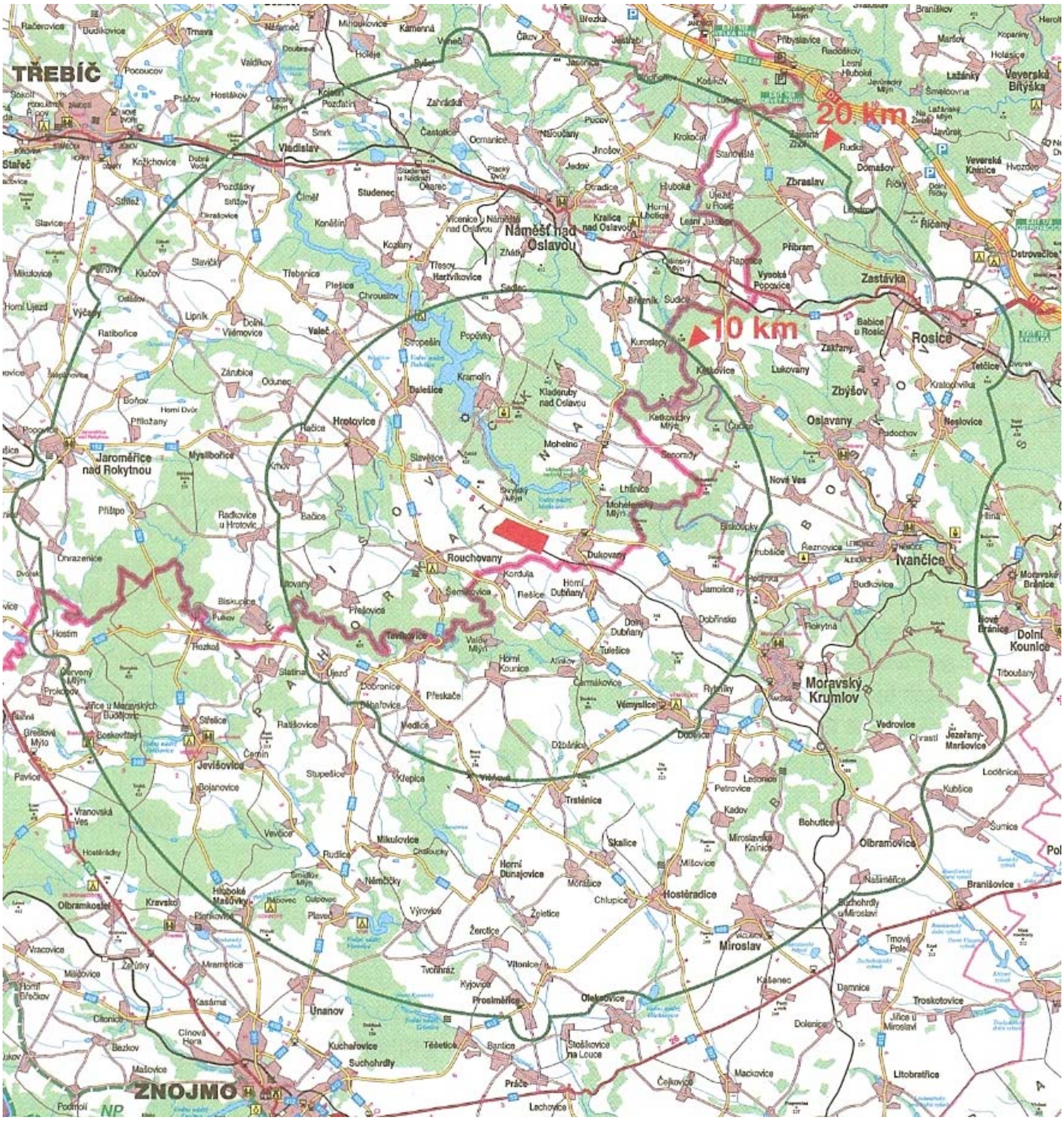


Fig. 18: Emergmergency planning zone of NPP Dukovany

Measures taken to enable optimum intervention by personnel

During a threat to safety on the unit or in the site or during the origination of a situation that cannot be managed by labor forces in the shift, the shift engineer announces three levels of extraordinary events.

EE 1 level (Alert),

EE 2 level (Site emergency),

EE 3 level (General emergency).

The procedure for evaluation of the seriousness of originated extraordinary events in individual power plants is stated in the respective action instructions. The seriousness of the originated reported events is evaluated by SE by comparing the type of reported event with the set of the previously defined intervention levels. The EE can also be classified by the commander of the emergency board. The intervention levels represent the set of previously stated, locally specified, initiation conditions when the status of the nuclear power plant is classified into the respective classification level and type. The intervention levels are prepared for all operating regimes of the nuclear power plant. The initiative condition may exceed any of the stated parameters, or the occurrence of discrete internal and external events that may threaten the nuclear safety and radiation protection of the nuclear power plant.

Types of extraordinary events

The timely identification of the type of originated event and evaluation of the seriousness in terms of nuclear safety enables selection of an appropriate response. Extraordinary events are divided into three basic types:

- Events from technological reasons
- Radiation events
- Events from other risks

This classification of intervention levels enables the system engineer to easily identify the seriousness of the originated extraordinary event in relation to ensuring nuclear safety and radiation protection.

The declaration of EE is in the full competence of the SE. In the case of declaration of EE 1 stage, only the technical part of SOER is activated (technical supporting group - TSC); in the case of the announcement of EE of the 2 and 3 stage, the remaining part is activated – Emergency response board NPP Dukovany (ERB).

SE is liable for all activities related to the resolution of EE from the time that the commander of ERB accepts responsibility for the EE solution. SE continues to be responsible for the management of technology in the undamaged units. The liquidation is managed by safety supervisor on the affected unit. SE is responsible for the fulfillment of the commands from the ERB commander in the area of the management and coordination of the activity of the work shift.

Up to the time that the commander of ERB takes responsibility, SE ensures:

- the provision of the stated information
- coordination of assistance of LFRU, medical readiness for action, guarding G4S
- in the case of EE level 3, issues warning to employees in the NPP Dukovany site and EPZ inhabitants
- manages the safe removal of personnel, if necessary

The workplace of TSC and ERB is the Emergency Control Centre (ECC), which is located in the NPP Dukovany site. In the case of the announcement of EE of the 2nd and the 3rd level, the following centres are activated:

- logistics support centre in Třebíč (concentration, boarding and accommodation of necessary experts for the solution to the emergency situation),
- emergency information centre in Třebíč (ensuring contact with journalists and keeping the public informed)
- external emergency support centre in Moravský Krumlov (ensuring radiation monitoring in EPZ). All these centres are established by the emergency board.

The organizational method of managing of extraordinary events is stated in the internal emergency plan approved by SÚJB.

For the solution of technological accidents (up to damage of fuel), strategies have been prepared that are contained in the emergency operating procedures (EOPs). To mitigate the consequences of accidents after damage to fuel (major accident), strategies have been prepared and are contained in the manuals for managing major accidents (SAMG). In EOPs the main priority is the restoration of the heat removal from RC and the prevention of damage to the 1st barrier against the release of fission products (covering the fuel), whereas in SAMG the main priority is the prevention of damage to the 3rd barrier against the release of fission products (containment), which at this time is the last integrated barrier.

Internal organization of emergency response
The internal emergency response team consists exclusively of shift personnel, i.e. employees who work on the normal operation of the NPP. Personnel from the permanent shift ensure, according to the instructions of the shift engineer, all activities related to the suppression of the effects of the originated extraordinary event up to the time of the activation of the on-call team of employees who are ready for action within the organization of the emergency response.

The shift engineer, in the case of the origination of EE, is responsible for the announcement of EE and the management of activities in accordance with EE up to the time when responsibility is handed over to the activated ERB commander. His activity during the origination of EE is managed according to ZI for SE which lists all responsibilities and authority, of which the most important are: evaluation of the seriousness of EE – classification, ensuring notification and warning of personnel of the NPP and warning in EPZ, notification of the management of the NPP and the respective bodies and organizations about the origination of EE, decision on the activation of SOER, decision on protection measures for the NPP personnel. The responsibility for technology continues to be in the competence of SE.

The personnel for permanent shift operation (with the exception of the personnel for the shift on CR) in the case of the announcement of an extraordinary event depending on the level of seriousness, either executes the activity according to the respective intervention instructions and the instructions of the management personnel of the shift, or are concentrated in the case of the declaration of protective measures in the operating support centre in the hiding place under the operating building, from where on the basis of SE of ERB they make the required interventions in the technology or create operative support for the LFRU unit during the recovery and rescue work.

For the requirement to ensure the implementation of protective measures for hiding and evacuation, teams are established which ensure the activation and the consequent operation of hiding places in the NPP site. The basic obligations of members of the hiding teams in the hiding places are: management of the regime in the hiding place, record keeping of hiding places, order service, operation of the air ventilation system, dosimetric measurement of per SÚJB servicing DGS.

Emergency organization of emergency response
The emergency response organization consists of personnel of the emergency supporting services ensuring week-to-week continuous preparation for action.

- Emergency board
  The emergency board is the main management workplace of the emergency response organization of the NPP. After activation it ensures the announcement of protective measures for employees and other persons located in the site of the NPP at the time of the origination of a extraordinary event, management of activity of all employees and other persons participating in the action during the suppression of the development and resolution of the consequences of an extraordinary event in the nuclear power plant and ensures communication with external elements of emergency preparation. The emergency board ensures the delivery of necessary material, special means, alteration of personnel and material ensured through the logistic support centre.

- Technical support centre
  The technical support centre is professionally staffed to provide qualified technical support for the personnel of the control room for the affected unit during the resolution of extraordinary events. At the same time, TSC ensures the immediate evaluation of the security status of the nuclear power plant with an emphasis on nuclear safety and radiation protection, manages the activity of the operatively stated action groups during the resolution of the consequences of extraordinary events and is able to prepare source materials and recommendations for making decisions and controlling the activity of the emergency board. The head of TSC may request through SE or the commander of the emergency board the strengthening of TSC by further experts.

- External emergency support centre
  External emergency support centre ensures activities related to radiation monitoring and evaluation of the radiation situation in the emergency planning zone and on the basis of the results of radiation monitoring and the prognosis of the further development of the radiation situation.

- Emergency information centre
  The personnel emergency information centres ensure, in the case of origination of an extraordinary event, the distribution of information to mass media and answering questions from the public. Its activity is to inform the general public and state administration bodies and self administration not involved in the system of external emergency, about the status of the nuclear power plant. It is responsible for the preparation of press releases for mass communication media. The emergency information centre is located on the premises of the Hotel Atom in Třebíč.

- Logistics support centre
  The personnel in the logistics support centre ensures the necessary means of technical material and highly-qualified staff according to the requirements and demands of the emergency board, technical support centre and external emergency support centre. The logistic support centre represents the external ERO support. The logistic support centre is located on the premises of the Hotel Atom in Třebíč.

Use of off-site technical support for accident management

Ensuring of external support and use of further capacities, resources and means is managed in ERB by the logistics employees, in cooperation with the logistic support centre.

For assistance with transport or heavy technology, the integrated rescue system of the Czech Republic will be used with the operating centre at FB for the Vysočina region, or FB of the South Moravian region which has the authority within IRS to call on further elements and organizations to help when managing the consequences of an extraordinary event. Within the whole ČEZ group, for the affected site, help is stated through the crisis board of ČEZ for the affected area. Within this body, access to external experts has been ensured (suppliers, experts from research institutes ÚJV Řež, EGP, VÚJE with knowledge of the respective issues, foreign help from other NPP of the VVER type in the localities of the power plants at Bohunice, Mochovce, etc.). The most effective help is proposed to be from Temelín NPP.

The external preparedness of the NPP is ensured with the participation of a series of bodies and organizations at both a national and local level.

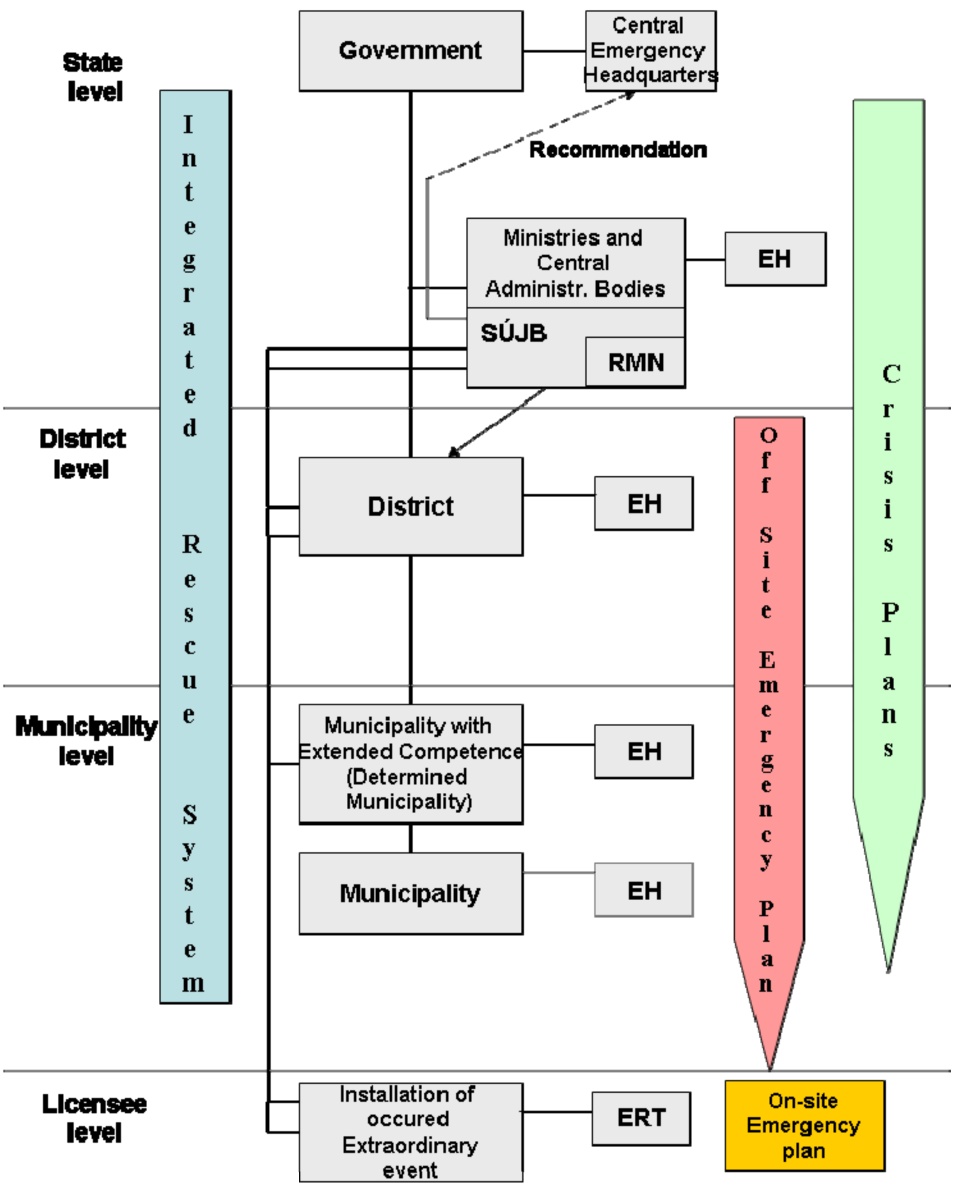


Fig. 19: Ensuring the external emergency preparedness of the NPP

During the occurrence of EE and the consequent resolution of the originated EE, the nuclear power plant communicates with external bodies and organizations at a national and local level.

- SÚJB – Crisis board
  The SÚJB crisis board ensures, through radiation monitoring of the network of the Czech Republic, an independent evaluation of radiation effects of the originated extraordinary radiation event. On the basis of the results of the monitoring of individual elements of the network of the Czech Republic, it provides source materials for decisions by the crisis board for the region on measures for the protection of inhabitants.

- Regional government
  The regional government ensures coordination of the external emergency preparedness of all villages with expanded authority where the territory intervenes into EPZ. The supervisor of the respective region, in cooperation with the mayors of the affected villages with expanded authority, manages all activities related to ensuring external emergency preparedness in the whole emergency planning zone and decides on the announcement and implementation of measures for the protection of inhabitants. The crisis board for the

region serves as an advisory body for the supervisor. The declaration of urgent protective measures is made on the basis of the recommendations of the SÚJB crisis board prepared from the results of radiation monitoring and other resource materials provided by individual elements of the radiation monitoring network.

The operator provides, in the case of a radiation accident in the nuclear power plant, for the crisis board of the region and through the emergency board, the necessary assistance, data and information required to evaluate the seriousness of the originated situation. To ensure assistance, the nuclear power plant will send a representative to the crisis board of the region.

- Villages with expanded scope of authority
  The mayors of affected villages with expanded authority will decide on the calling up of the crisis board for their village and manage the announcement of protective measures in the concerned territory of the village with expanded authority. During the management of these activities, they use the external emergency plan. Protective measures are announced, after prior discussion with the crisis board of the region, which ensure the coordination of messages and information passed between individual villages with expanded authority, SÚJB and the nuclear power plant. This procedure serves to ensure the line of announced protective measures in the territory located under the management of individual villages with expanded authority.
- Czech Hydrometeorological Institute
  The Czech Hydrometeorological Institute evaluates for nuclear power plants the actual hydrometeorological situation and prepares a prognosis for further development. The outputs of basic hydrometeorological data necessary for the evaluation of the potential or actual spread of radioactive releases in the surroundings of the nuclear power plant are handed over to the respective information networks of the nuclear power plant.
- IRS – Integrated rescue system
  The integrated rescue system (hereinafter referred to as IRS) exists for the purpose of the coordinated management and the solution of extraordinary situations on the basis of detailed specification that it concerns industrial discrepancy, floods, earthquakes or other natural disasters. In terms of legislation, this issue is resolved in the laws on integrated rescue systems and crisis management. Within IRS, a central IRS alarm plan is prepared which will be used when necessary, due to the extraordinary event of the crisis situation or a safety event, and when the legally stated conditions are fulfilled for the central coordination of rescue and liquidation work, or if the regional supervisor, the mayor of the village with expanded authority, director of FB of the region or the commander for action required through the ITS operating and information centre of the region for help and labor forces which IRS units do not have at a regional level the means for performing rescue and liquidation work during the extraordinary event to be resolved independently in the respective region.

  Labor forces and the means for the central coordination of rescue and liquidation work are called and implemented by the Ministry of the Interior – General Directorate of the Fire Rescue Brigade of the Czech Republic (hereinafter referred to as the "General Directorate") through its operating and information centre.
- Fire rescue brigade
  The fire rescue brigade, on the basis of instruction from the nuclear power plant, is responsible for warning inhabitants in the emergency planning zone using sirens controlled through the national integrated warning system and also ensuring broadcasting of the respective public media. On behalf of ČEZ, a. s., FB also notifies the concerned villages with expanded authority through the regional operating and information FB centres (in accordance with Regulation No. 318/2002 Coll., as amended).

- Police, security service and army
  Within IRS, among others, 6 helicopters are reserved for rescue work (Czech Army and the Czech Police) with the possibility to transport persons and loads where 4 crews are in the ready for operation status with the possible activation within 10 minutes during the day and 20 minutes during the night.

- Rescue medical service (Traumatological plan)
  In NPP Dukovany premises, on a contractual basis, a medical first aid service is established with continual readiness for operation. The document includes relations and the flow of information during reaction to a traumatological event in the case of the origination of an extraordinary situation and its relation to authorized medical facilities (four “Centres of specialized medical care” are to be established).

Procedures, training and exercises

The concept of managing technological accidents at NPP Dukovany is based on symptomatic access. At present, the following strategies are prepared for NPP Dukovany for the solution to the beyond design basis and severe accidents.

- Symptomatically oriented emergency operating procedures (EOPs) for power modes
- Symptomatically oriented emergency operating procedures for shutdown modes, including cases of threats to the heat removal from the spent fuel stored in SFSP (SDEOPs)
- Manuals for decision making by TSC
- Severe Accident Management Guidelines (SAMG).

All the mentioned procedures and manuals were developed and are updated in cooperation with the Westinghouse company.

The procedure to treat an emergency situation beyond the scope of EOPs is in accordance with the internal emergency plan (announcement of the level of an extraordinary event).

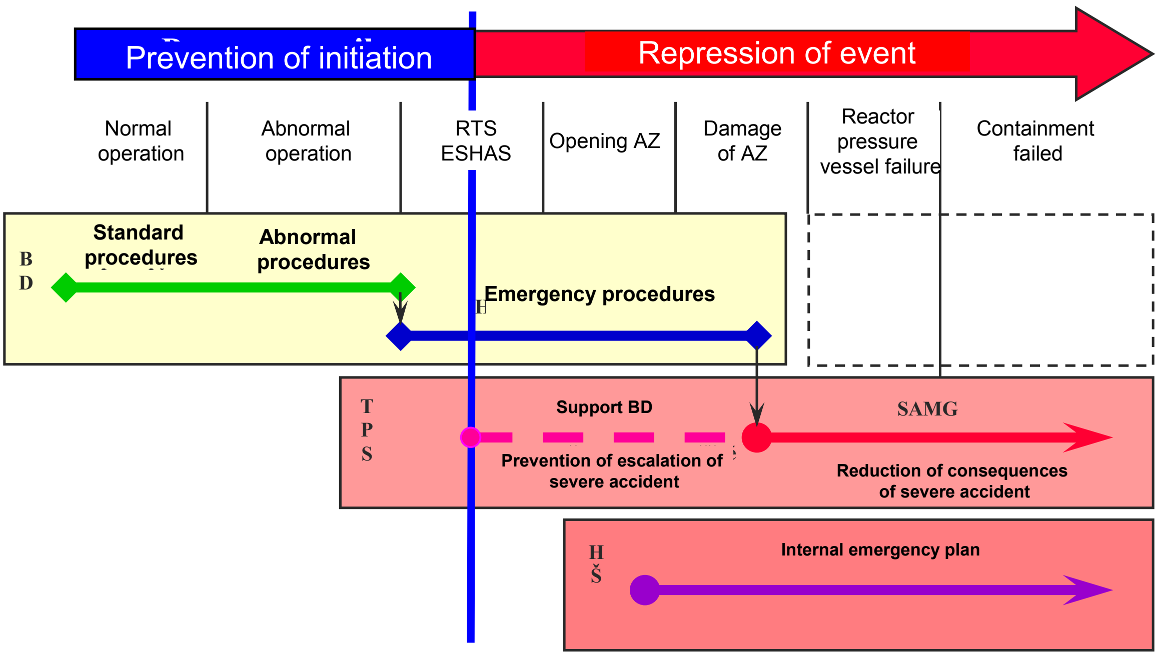


Fig. 20: Relation between the status of the unit, operating documentation and EE

The activities of the operative personnel at each level are managed by the operating procedures adapted to each operating status. The procedures represent the network which states the activities of the operating personnel in each operating status of the unit.

For emergency conditions in the preventive phase, strategies are prepared that are included in the emergency operating procedures (EOPs). For managing severe accidents, strategies are prepared which are included in the Severe Accident Management Guidelines (SAMG). The basic condition for executing the activity according to emergency procedures is that the status of RC enables cooling, i.e. RC is a geometric configuration that can be cooled. In the case of irrevocable damage, emergency procedures need not provide the optimal manual for the solution of the emergency situation and it is necessary to proceed according to SAMG. At this time, the main priorities are changed. In EOPs the main priority is the restoration of the heat removal from RC and the prevention of damage to the 1st barrier against the release of fission products (covering the fuel), whereas in SAMG the main priority is the prevention of damage to the 3rd barrier against the release of fission products (containment), which at this time is the last integrated barrier.

The objective of interventions described within EOPs, which the CR operating personnel will use for the solution of the design and beyond design basis events, is to ensure sufficient cooling of RC and prevent irrevocable damage to RC and also minimize the consequences of any release of radioactive substances outside the NPP. The philosophy of these actions includes the continuous evaluation of critical safety functions monitoring the statuses of physical barriers against loss of activity. This evaluation ensures the timely identification of the worsening of the safety status of the unit and ensures timely correction in the case of ascertaining a negative trend in the development of the event.

The set of symptomatic oriented emergency operating procedures provides a systematic tool (independent to the course of the emergency regime) for operative personnel to report emergency situations by means of a set of previously stated and structured emergency procedures. The combination of event and function oriented strategies provides operating personnel with guidelines for putting the unit into safety and end status while ensuring continuous diagnostics of the status of the unit and the restoration of the safety status independently of the course of the stated emergency event.

The emergency procedures also contain systematic means for the evaluation of the safe status of the unit through the evaluation of the statuses of critical safety functions. The critical safety functions closely relate to the physical barriers which prevent the release of radioactivity into the surrounding environment.

For the preventive phase of managing the emergency situation when the operating personnel does not proceed according to EOPs, the TSC personnel has manuals available (“Manuals for TSC”) containing source materials for making decisions during the support of operating personnel executing activity according to emergency procedures. The emergency procedures include many steps that explicitly require instructions for further activity from TSC personnel. At the same time, experience from training, simulation on a full-scope simulator, etc., shows that support from TSC personnel is required in a series of other situations that do not explicitly require it. In all such cases, the decision depends on the actual development of the emergency situation and the specific status of the systems and equipment in the units.

These manuals were created for TSC personnel and for further NPP personnel who, in addition to TSC personnel, are entitled to provide support for decision making.

- Manuals for TSC are used by TSC personnel; when MU1 has been announced, TSC has been called and TSC personnel are able to provide support.
- Manuals are used by safety supervisor, SE or RUS, if support for a decision is requested before then, TSC becomes functional.

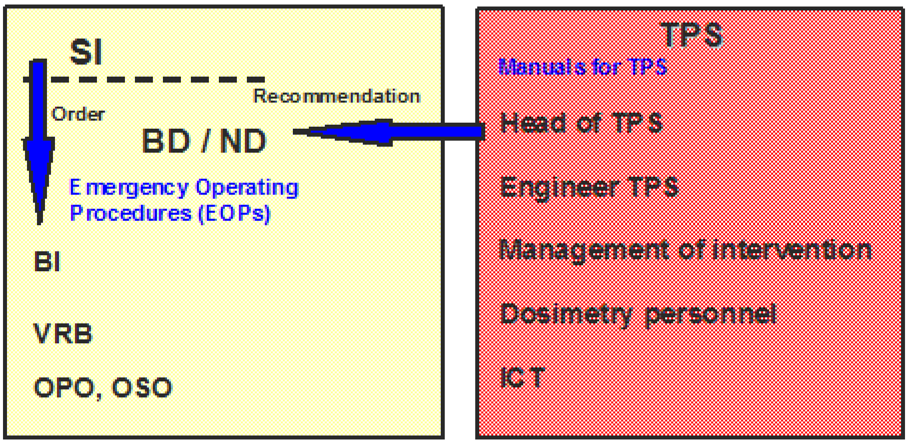


Fig. 21: A diagram of communication between TSC and the operating personnel when using the Manuals for TSC

In the case of the development of an event in the area of the severe accident, a further procedure is selected to at least ensure the remaining barriers against the release of radioactivity. The loss of integrity and geometry of the fuel in RC means a serious threat to the ability to remove heat from RC. Under these conditions it is not possible to proceed according to EOPs. SAMG for achieving stabilized status are prepared for this phase of the accident.

The transfer to SAMG takes place in the case that irrevocable damage to RC is ascertained. In this case activities according to EOPs are terminated and the transfer to SAMG takes place. The only input point into SAMG is the manual SACRG-1, ACTIVITY CR WITHOUT TSC.

There are three possible transfers from EOPs into SAMG:

FR-C.1 Loss of cooling RC

FR-S.1 No shutdown of reactor (ATWS)

ECA-0.0 Loss of electricity supply – Blackout.

These three transfers for emergency statuses into SAMG are sufficient and cover all possible severe accident scenarios.

To decrease the consequences of severe accidents, the following objectives must be fulfilled:

- **Primary objectives of SAMG**
  - Restoration of the heat removal from RC or from melt = restoring the source of the development of heat to a stable status that can be controlled.
  - Keeping the integrity of the containment as the last barrier against the release of Ra substances into the surroundings = ensuring the status of containment that can be controlled
  - Terminate the release of Ra substances into the surroundings
- **Secondary objectives of SAMG**

– Minimizing the release of Ra substances while fulfilling the primary objectives

– Ensuring the maximum operating capability of equipment during the fulfillment of the primary objectives

For managing severe accidents, the symptomatic oriented approach is consistently applied. The basic principle of this approach is that the respective strategy of the solution is selected on the basis of the actual development of the accident which is identified on the basis of unique symptoms (features). If during the solution of the accident there is a change in the symptoms and the chosen strategy cannot be used, then the structure of procedures and manuals enables changing the original strategy and continuing with activities stated by another procedure or manual which better corresponds to the originated conditions. The continuous diagnosis of the status of the unit during the accident enables correct response to changing conditions in the development of the accident and interventions are always an optimal response to the status of the unit and also take into consideration the external event and the risks.

In the case of activity according to emergency procedures being terminated and the transfer into SAMG, activities according to the manuals for TSC are terminated and for further management of the activity, only SAMG are used.

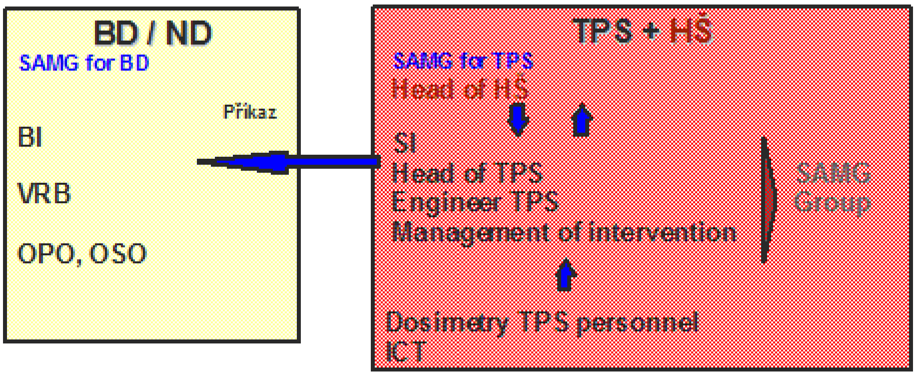


Fig. 22: Diagram of communication between TSC and operating personnel during the use of SAMG

To date there are no manuals for NPP Dukovany for managing severe accidents for shutdown states (SAMG for shutdown states) designated for beyond design basis accidents due to damaged fuel in RC, which develop into a severe accident during the shutdown of the reactor (open reactor) or for the beyond design basis accidents due to damage to fuel in SFSP.

For maintenance EOPs, SDEOPs and SAMG maintenance there are regular updates that include knowledge from their use on the simulator or during emergency drills. External knowledge (within the “users group” and long-term cooperation with Westinghouse) is included in this documentation in the form of the “Maintenance program”.

For the relevant workplaces for the shift and supporting personnel and for execution of activities when managing accidents, personnel and qualification conditions are stated and these requirements are also checked through the set of qualification preconditions. For each workplace there are requirements for education, specific knowledge (basic preparation, periodical preparation, simulator training, training of selected personnel in the area of the beyond design basis and severe accidents) and professional development training. A system of requirements for qualification is introduced for selection of SOER personnel, and further criteria taking into consideration their knowledge and professional orientation are considered.

The preparedness of the shift and technical personnel for managing technological accidents (including the transfer from EOPs to SAMG) is regularly checked during training on the full-range simulator with the participation of TSC personnel (2 per year) and during emergency training.

Emergency exercises take place a minimum of 4 times a year so that each SOER shift passes the exercise at least once a year. The exercises also include preparation for versions of preventive actions in severe conditions. The respective procedures were prepared for the activities of the intervention groups in severe conditions and for their protection. The real drill in the use of SAMG when managing severe accidents on NPP Dukovany took place after putting SAMG into use (under the supervision of Westinghouse).

As the full-range simulator is not designed for a simulated course of severe accidents, a new simulating tool is being developed with UJV Řež enabling the display of courses of the parameters and their behavior in time and space. It is one of the measures resulting from the Periodical Safety Review. This simulation tool will be used during the creation, update and maintenance of SAMG and, at the same time, will also be used for training the personnel of Dukovany NNP (mainly TSC) in the use of SAMG procedures for managing severe accidents. The tool is based on an animated display of the course of a severe accident in the reactor, the primary circuit and the containment. The display will be interactive: according to selection, it is possible to change the speed of the display, repeat selected sections of the accident and select the set of additional animated graphs of the characteristics of the accident.

Dependence on the functions of other reactors at the same site

In the NPP Dukovany site there are four reactor units arranged into two twins. Containments of the individual units in the twins are strictly separated during the operation and there is no risk of penetration of the atmosphere from one unit into the other. In modes 6 and 7 for the replacement of fuel on one unit, the containment is opened into the reactor hall which is shared with the neighboring unit; however it is hermetically separated from the containment of the neighboring unit. In the reactor hall there are also pools for storing fuel for both units. In the case of an accident when replacing the fuel, it is necessary to resolve the issue of Ra substances spreading into the common reactor hall and the opened containment of the affected unit.

Reactors are technologically completely independent; nevertheless a series of systems and auxiliary and supporting systems are mutually used. For example, electricity supply, circulating cooling water, fire water, etc., systems are connected between each unit.

Essential service water (ESW) for example, has a similar property. There are six independent systems on each unit that cool the key consumer appliances. Pumps for connecting individual units are located in separate cubicles in one building, their electricity supply is from the respective units and systems and they can also be used between neighboring units, i.e. units 1 and 2 are serviced by common pumps, similarly units 3 and 4.

The twins arrangement of auxiliary systems enables, in an emergency status, the replacement or adding of media to the tanks for the emergency systems (ECCS) from the neighboring unit. It is also possible to use, if only one unit from double units is damaged, reserves of water in the XL passive emergency system for the neighboring unit, which can represent a minimum of 1000 m^3 of the solution H_3BO_3.

Due to the total of four units in the site arranged in the neighborhood and the double unit system and due to the independent electricity supply for individual units from external and internal sources (including emergency), it is possible to use resources from the electricity supply source for one unit during the origination of SBO on another unit

The folding pools are located in the reactor hall at each reactor. The reactor hall for the double unit is linked. Due to the common support systems (system for cleaning TM water) it is possible to link the units in the case of emergency filling. The location of the folding pools outside the containment enables simple access in the case of emergency filling by other emergency means (extinguishing technology, etc.).

II.6.1.2 Possibility to use existing equipment

Provisions to use mobile devices (availability of such devices, time to bring on site and put into operation)

For ensuring the safety functions (for the design basis and beyond design basis scenarios), the respective symptomatic oriented procedures and manuals are prepared (EOPs or SAMG). In EOPs and SAMG the use of mobile LFRU means in the site are utilized. Other mobile or non-technological means from external sources, with the exception of the use of further extinguishing technology from nearby fire brigade units, are not currently being considered.

In the NPP Dukovany site there is an available fire brigade unit for the enterprise (LFRU), which has the respective fire technology and is trained for action in any part of the site. The fire technology and action personnel (48 fire fighters in 4 shifts) are located in the fire station where there is no risk of the direct effects of extreme natural phenomena; however, the seismic resistance of this object has not yet been evaluated. In the case of damage, the access of the fire brigade could be restricted.

For the demands of strengthening LFRU Dukovany NPP, an alarm plan is prepared which is part of the external plan of Dukovany NPP on the basis of which the professional units of the FB of the Czech Republic will be able to ensure further effective material and personnel help which are parts of the IRS with arrival to the site within 10-60 minutes depending on the location of the fire unit.

The pumping technology of LFRU is one of the main mobile non-technological means that can be used to transport and pump the media. It is also adapted for pumping water in the case of floods.

Directly in the site are three mobile LFRU pumps (the pressure on delivery of the pump is 0.8-1.2 MPa, flow 120-150 t/h), which can be used simply to add demiwater directly into SG as an alternative. Within the completion of the project connection points are prepared that enable the fire technology to be connected with the other technology. The alternative manner of completion of SG as described in EOPs was practically tested several times and the capacity of this technology was checked for ensuring of basic safety functions. The real time for the actual delivery of water into SG by the mobile pump from the time of the request for activation of LFRU is about 20 minutes. In the case of the loss of measurement of water level in SG and other data for the possibility of optimally adding demiwater, tables are prepared where the flow of demiwater into SG is necessary with the respective counter-pressure in the SG so that the flow of added demiwater corresponds to the collection of steam through SBSA. In the case of black-out status on all four NPP Dukovany units at the same time, a certain restriction may be the capacity of the necessary fire technology (emergency plans for the supply of SG for two units with one pump at the same time are not prepared yet).

An alternative is to use fire technology for adding spent fuel and keeping the temperature of the fuel in SFSP. EOPs mention this alternative adding of SFSP; the actual procedures for interventions in the area have not been prepared yet.

In addition, in SAMG use of portable diesel aggregates has been considered to control some valves directly from distributors; the actual procedures for intervention at the location are not prepared yet.

In addition, according to the respective legislation, it is possible to use basic and other elements of the IRS (NPP medical centre, Police of the ČR, Army of the ČR…). According to the level of the extraordinary event on the nuclear equipment, the individual elements acting in the rescue systems fulfill the tasks aimed at the liquidation of an extraordinary event on the damaged equipment to restrict the consequences. Tasks may be performed in the territory of Dukovany NPP, inside or outside the emergency planning zone.

Provisions and management of supplies (fuel for diesel generators, water, etc.)

The fuel reserve in the operating tank for each DG is for a minimum of 6 hours (4.5 m^3 fuel, consumption at maximum loading of 0.7m^3/h). For each DG there is one mutually linked pair of storage tanks where there is a minimum reserve of 110 m^3 fuel. The pumping of diesel from storage tanks into the operating tank is automatically performed upon the lowering of the level in the operating tank. Pumps for transporting fuel have an electricity supply from the respective DG. The total 114.5 m^3 diesel reserve is sufficient for the operation of one DG for at least 144 hours (actual is about 160 hours), i.e. for 6 to 7 days without the need to add external fuel.

Further fuel for DG could be gathered by re-pumping from other DG (which are, for example, out of operation), the re-dispatching pumps under condition of gathering an electricity supply for their operation (from an unsecured system). When considering that for long-term operation there is only one DG on each unit, then when commissioning the pumping of reshipped diesel, there is fuel for 18 to 21 days without the external supply of fuel to NPP Dukovany. The quality of the diesel is regularly checked and preventively changed.

For the completion of demiwater by the optimal flow, the existing reserves of demiwater are available for tanks 3 x 1000 m^3 for each double unit, which, according to the analysis, is sufficient for 72 hours for all 4 units. Together with the use of the reserve of coolant in FWT, adding water into SG of all four units of the NPP, there is a reserve of coolant for about 4 days. In addition to the reserves of coolant in the demiwater tanks, for the supply of SG by mobile means, it is also possible to use the cooler from the cooling towers pool or other sources.

In the case of the loss of adding raw water, if there is an unsecured electricity supply available it is possible to use the cooling reserve in clarifiers of about 5 x 2000 m^3 and reserves of raw water and gravitation water tanks with a volume of 4 x 2000 m^3 for compensation for losses of ESW by evaporation.

It results from the pumping station Jihlava and ESW analysis that in the NPP Dukovany water systems, in the case of a conservative approach (only half CPS I and CPS II is considered, the level in cooling towers at min. value -2.55 m) about 75 564 m^3 water is available. This reserve is sufficient for 931 hours (about 39 days) for the collection of residual heat (operation of ESW pumps) from shutdown reactors without adding water into the NPP Dukovany systems.

Management of radioactive releases, provisions to limit

The objectives of all strategies for managing accidents (in EOPs and SAMG) are aimed at preventing threats to the health and safety of persons. If, despite this, during an accident there is an release of radioactivity, then the whole effort when managing the accident is aimed at terminating or at least restricting these releases.

The Ra situation inside and outside the site is monitored through the central radiation monitoring system (CRMS), which in the existing design does not have the seismic qualification at the level of the maximum design earthquake and is located in premises that do not have this seismic resistance to earthquakes with the intensity > 6° MSIS-64 (PGA_{hor} >

0.05g) and also do not have a supply from a 1st category secured supply. For measuring the radiation there will be an alternative manner available by using portable measuring devices.

For the prognosis of the consequences of an release of the radioactivity and for the evaluation of the actual radiation situation in the case of an release, NPP Dukovany uses RTARC software, which uses prompt meteorological data, predicted meteorological data, data about releases, dimensions of MPU, data about the relief of the terrain of the surroundings of the NPP and data about radio nuclides represented in the released radioactive substances. The output is the actual radiation situation and its prognosis for the selected period in the surroundings of the NPP from 500 m to 40 km.
The RTARC system operates with data in the previously calculated source member which can be corrected on the basis of the actually changed release values. For the time being there are undeveloped source members taking into consideration the melting of the fuel in the open reactor and in SFSP (storage pool for spent fuel).

The restriction of radiation for persons and the environment during an extraordinary radiation situation is performed by protective measures, which are:

- Urgent measures which include hiding to safe area, iodine preventive dose and evacuation.
- Consequent protective measures that include relocation, regulation of the use of food and water contaminated by radio nuclides and regulation of the use of feed contaminated by radio nuclides.

The protective measures during radiation accidents are always carried out if they are justified by a greater contribution than the costs for measures and damage caused by them and they are optimized concerning the form and duration so they bring the maximum reasonable achievable contribution.

Depending on the level of the radiation situation, in the case of the announcement of an extraordinary event, the respective urgent protective measures are announced to ensure that the intervention levels mentioned in the following table will not be exceeded where there are radiation limits for employees and other persons for the announcement of protective measures in the case of the origination of an extraordinary event within eight hours after the origination of the event.

Tab. 20: Limits for the radiation of persons in the case of the announcement of an extraordinary event on NPP Dukovany

	PROTECTIVE MEASURE		
CATEGORY OF PERSONNEL	**Hiding**	**CATEGORY OF PERSONNEL**	**Evacuation**
Other persons and employees not classified into ERO	5 mSv	Other persons and employees not classified into ERO	5 mSv
ERO personnel	50 mSv	ERO personnel	50 mSv
ERO personnel in the case of saving lives or preventing the development of an extraordinary radiation situation.	According to Section 4, par. 7, letter c, of Act No. 18/1997 Coll.	ERO personnel in the case of saving lives or preventing the development of an extraordinary radiation situation.	According to Section 4, par. 7, letter c, of Act No. 18/1997 Coll.

The above mentioned limits do not relate to the radiation of persons participating in interventions in the case of a radiation accident, however, this radiation must not exceed a multiple of 10 of the basic limits for employees with resources (the value of the basic limit is

50 mSv per calendar year or 100 mSv during 5 consequent calendar years) if it is not the case of saving lives or preventing a radiation accident with potential wide ranging social and economic consequences. Employees carrying out the action are familiarized with the risk and expected level of the received dose prior to the action.

There are 7 shelters in NPP Dukovany. In one shelter is ECC, in one is the operating supporting centre where, among others, the shift personnel would be concentrated that are necessary for carrying out local actions. The capacity of the shelters is 2,450 persons. Each employee in the workplace or in the shelter has personal work aids for protection against surface and internal contamination (protective clothing - TYVEK overalls, covers for shoes and one dose of KI iodine prevention).

Iodine prevention by taking potassium-iodide tablets is available for all persons with the exception of persons older than 45 for whom sensitivity to iodine products was previously proven or who have thyroid gland defects. All NPP employees and EPZ NPP Dukovany personnel are supplied with potassium-iodide tablets. EPZ NPP Dukovany personnel are given a manual for the protection of persons which contains instructions in the case of a radiation accident. It includes instructions during warnings by sirens, instructions for primary protective measures (shelters, iodine prevention and evacuation).

Communication and information systems (internal and external)

List of internal means of communication:

- System of warning and notification of personnel (external sirens, internal sirens, intra-copmany broadcasting)
- Telephone switchboard
- Local communication equipment
- Radio networks, freeset system
- Paging system
- Radio network amplifiers (radiating cables)
- Portable radio station
- Mobile radio stations
- Communication system for calling SOER

List of external means of communication:

- System of warning and notification of inhabitants (sirens in EPZ)
- Prepared records for state communication media (television, radio)
- Network of the telecommunication operator O2 (mobile and fixed)

Periodical verification and inspection is carried out for all communication systems.

- Every three months - the functionality of hardware, systems and methods of activation of persons intervening and performing actions for the management.
- Every six months - the functionality of the technical means of the systems and methods for warning employees and other persons in the NPP.
- Once in three months - the functionality of hardware, systems and methods for announcing the extraordinary event and the notification of the radiation accident.
- Twelve times a year - the functionality of hardware, systems and methods of warning inhabitants in the emergency planning zone.

For activating the members of the emergency response organization and for activating sirens, there are a minimum of two methods (autonomous and independent for potential overloading of mobile networks). For selected mobile telephones, through the IRS operating centre, the priority for calling within the network during the solution of an extraordinary event is set. NPP Dukovany has an independent system that is used to activate sirens and to notify the members of the emergency response organization. The system is autonomous and does not depend on any overloading of the mobile network.

The emergency control centre contains an information system that ensures access to all information necessary for the management of extraordinary events. In the case of the origination of an extraordinary event, there is audio contact from TSC with all CR. TSC personnel have actual on-line technological data and radiation data available which is also used by the operative control personnel. The project for the visual camera system CR, TSC, SE workplace is in the stage of completion, including the completion of industrial cameras.

II.6.1.3 Evaluation of factors that may impede accident management and respective contingencies

Extensive destruction of infrastructure or flooding around the installation that hinders access to the site

Access to key premises could be restricted due to the destruction of non-seismic resistant buildings on the internal access road, as well as debris in the entrance to the power plant. In this case the alternative entrance into the site could be used.

Loss of communication facilities / systems

The backup supply for operation of the means of communication for warnings in the site, as well as for the connection of key personnel (ECC, shelters, LFRU, SUJB, IRS, CR personnel) is ensured in the case of loss of the supply or damage to the infrastructure, usually within several hours. Sirens in NPP Dukovany buildings do not have a backup supply. The intra-company broadcasting is without a backup supply. Sirens in buildings have their own accumulator batteries. The operating radio has a backup supply.

In the case of longer SBO there could be the loss of the supply to telephones in the NPP Dukovany exchange and the telephone exchanges in cooperating workplaces outside NPP Dukovany, with the exception of the main control workplace at ČEPS Prague and the backup control workplace ČEPS Ostrava, which have their own DG. This threatens the restoration of the supply from the external network resources.

The restoration of the supply from resources outside of NPP Dukovany (e.g. from HPS Dalešice, or HPS Vranov) is conditioned by the cooperation (connection is necessary) of various external subjects (ČEZ, ČEPS, E.ON).

In the case of damage to the infrastructure, there can be a threat to the communication between acting persons and control centres, as well as to the external centres of state administration bodies (CCC SÚJB, crisis board of the region, IRS, etc.), because the accessibility and duration of the existing means of communication is very restricted. The fixed telephone network, mobile telephone network, transmitters, warning means, etc., are not secured against major damage to the infrastructure. Communication is available through LFRU transmitters to the other part of IRS (fire brigade in Třebíč).

The intra-copmany broadcasting at SBO is out of operation (completion by UPS station is planned). The operating radio at SBO remains in operation. Internal warning systems: internal sirens (located in buildings) will be in operation using their own accumulator batteries. The building sirens will be out of operation (the solution is to replace rotary sirens with electronic ones with accumulator batteries).

The transmission equipment between individual exchanges in the ČEZ ICTS network is dynamically controlled (selection of automatic systems through the free transmission route) and for this reason the time of its operation from the backup sources depends on which part has a loss of supply.

Active elements of the Duknet network are mostly supplied from switchboards (including user PC) and supported by UPS. The central node in administrative building 1 is designed for a period of 2 hours.

The MPLS WAN ČEZ network, which ensures the connection between data centres and individual localities of ČEZ, is backed up for 1 hour of the operation in the case of the loss of supply.

The control system of the internal warning system is in two independent workplaces for the operator of the electric control room and ERB, which are backed up from UPS.

Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some on site facilities

In cases of NPP damage there are no prepared crisis plans prepared for the use of supporting or alternative hardware. Any use of supporting and alternative hardware would be resolved by ERO mechanisms. If it is not possible due to any reason to use the emergency control centre, the alternative centre is the environmental radiation monitoring laboratory in Moravský Krumlov, in which a restricted volume of information for resolution of the extraordinary situations is available.

In the case of non-accessibility of the power plant, the situation would be resolved by the restriction of changes of personnel, with sleeping facilities in the site or nearby (in safe places and ECC, possibility of using the information centre building).

The personnel of the permanent shift operation, in the case of the announcement of an extraordinary event, depending on the level of severity, will continue to carry out activities according to the respective action instructions and commands or would be concentrated in the case of an announcement of protective measures in a safe place from where, on the basis of instructions of IS or ERB, they would perform the requested actions to the technology or create operative support for the LFRU unit during recovery and rescue work.

Each safe place in the NPP contains equipment enabling the protection of persons against the effects of radioactive substances, poisonous substances and biological means. In terms of the viewpoint of the construction they are designed so that they provide protection for people against the effects of penetrating radiation. The technical equipment in the safe place enables its operation for a minimum of 72 hours (including meals, drinks and hygiene). Basic equipment includes dosimetric devices for measuring surface contamination and dose input, reserve of spare emergency protective means, alternative clothing, means of iodine prevention, means for connection with the ERB workplace. The distribution of alternative emergency protective means, alternative clothing and medical material is managed by the members of the “safety team” on the basis of the justified demands and requirements of the people being protected.

There is no available heavy technology in the site for removing debris from backbone and access roads which could be demolished by debris from non-seismic resistant buildings. This could make access to the mobile technology in the main production units more difficult. There is connection through IRS.

Impact on the accessibility and habitability of the main and secondary control rooms, measures to be taken to avoid or manage this situation

The CR and ECR are located in rooms neighboring the containment. This part could be affected by radiation in the case of higher pressure and, at the same time, higher doses inside the containment or in the case of large fission releases from the containment. The fitting of CR and ECR with filter air ventilation systems has not been finished yet. The transfer of CR personnel to ECR is decided, in special cases, by SE or safety supervisor or RUS. The use of respiration devices at CR is in the competence of RUS.

Before termination of resisting CR, ECR it is necessary to consider the temporary evacuation on the basis of the command of the ERB commander on the basis of the creation of the radiation situation during the fulfillment of criteria in the action instruction (then it could be considered only short-term entrance for carrying out actions).

Impact on the different premises used by the crisis teams for which access would be necessary for management of the accident

In the case of the origination of an extraordinary event, all necessary activities must be managed and performed from protected places. Activities according to SAMG are managed by members of TSC and ERB, concentrated in the safe place designated for the emergency control centre. Remote activities for the implementation of the strategy would be performed by the control personnel from CR or ECR. The local actions and repairs of equipment would be performed in the respective rooms of the reactor, machinery room or external objects by the personnel concentrated in the operating supporting centre which is located in the shelter in the NPP Dukovany site.

Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods)

NPP and ERO personnel, in the case of earthquakes and floods, are qualified and trained for the use of EOPs and SAMG. However, they are not sufficiently trained for using the supporting and alternative technical means for which there are not the respective procedures, manuals and plans. Within the shift (ERO) or SOER no shortage of personnel necessary for the mitigation consequences of the beyond design basis events has been identified.

In the case of major damage to the infrastructure and the long-term inaccessibility of the site (destruction of building, damage to roads, etc.), the personnel would have difficulty accessing the site. In this case, the required activities should be ensured by the personnel that are present at the time of the origination of the event. The exchange of personnel would be operatively resolved in assistance with the state administration body (IRS, army, etc.).

Probably, it would not be possible to use the safe places of the emergency preparedness or workplaces of the emergency board or the technical support centre which are under seismic unstable buildings and are not protected against floods. The activity of TSC and ERB was resolved in this case operatively (there is no detailed information available).

The accessibility of information regarding the Ra situation inside and on the border of the NPP Dukovany site could be made more difficult. All existing systems for radiation control (CRMS) are not seismic versions because their parts are located in the non-resistant seismic premises of the operating buildings. In the case of the loss of information, the radiation would be measured in restricted range in the alternative manner using manual measuring instruments.

In the case of floods and long-term inaccessibility of the site, the incoming shift personnel need not operatively access the site. In this case, the required activities should be ensured by the personnel that are present at the time of the origination of the event. The changing of personnel would be operatively resolved in assistance with the state administration body (IRS, army, etc.).

It is also necessary to take into consideration the local restriction on using the roads in the surroundings of the NPP due to flooding or mudflows from the attached fields. The network of access roads and bridges through water flows in valleys will be enabled.

In the case of non-accessibility of the power plant, the situation would be resolved by the restriction of changes of personnel, with sleeping facilities in the site or nearby (in safe places and ECC, possibility of using the information centre building).

Unavailability of power supply

The restricted capacity of the SSPS I category accumulator batteries need not enable all actions in the early phase of a severe accident and could disconnect some measurements. One problem can be the long-term supply of key consumer appliances, including measurement necessary for SAMG. These measurements are concentrated in PAMS, which is supplied from the SSPS I. category. The acting personnel and personnel participating in the management and manipulation would not need to have all the necessary information.

Within SAMG it is considered as a possibility to use LFRU portable electricity stations to control some of the drives directly from switchboards.

Directly in the site, LFRU has available three electric stations 3x 380, 220V:

GEKO BSKA (5.5 kW), MITSUBISHI 4200 (3.6 kW). FORMULA 6000 T (3.6 kW)

Up to the time of fully discharging the accumulator batteries, the emergency lighting would be functional. The loss of lighting could contribute to more difficult orientation of personnel and to prolonging the time needed for manipulation.

Potential failure of instrumentation

Most of the requested information about the status of components and values of the parameters necessary for managing severe accidents is available in PAMS.

All systems are qualified for desigh basis and post-desin basis conditions. They are not qualified for conditions of severe accidents, but in most cases, their measuring range considers requirements for managing the initial phase of a severe accident.

- Temperatures from the output of RC up to 1200 °C.
- Temperatures in loops up to 400 °C.
- Overpressure in the box 450 kPa.
- Measurement of the concentration of hydrogen up to 10 %.

For the diagnosis of the emergency status and the verification of the implementation of selected strategies, a restricted set of parameters is used. For the verification of these parameters, the measured values of selected variables from the standard instrumentation are used. For each parameter, several values are stated where it is possible to verify the stated parameter (value, trend). Direct measurement of the requested parameter is always used and one or more measurements of alternative values on the basis of which it is possible to derive the value or the trend of the requested parameter. In some cases, it is not possible to evaluate for each accident the value or trend of the requested parameter on the basis of directly measured values either due to inaccessibility or non-existence of the measurement of the stated parameter. In these cases, for the determination of the requested parameter, calculating aids are used (simple graphs of dependences of parameters). Inputs from these calculating aids may be used either directly in measured values or from previously defined values.

The ability to restore measurements in the conditions of the environment for severe accidents is not known, although it is expected that they are sufficiently robust to resist, at least for a certain period, conditions during a severe accident.

The measurement of the concentration of hydrogen depends on the content of oxygen in the measured space and will be not exact in the case of a high concentration of hydrogen or consuming oxygen during burning. Despite this fact, it provides basic information on the risks of the hydrogen if, due to the low oxygen content, the measured concentrations will be significantly lower than the real ones; at the same time it means a very low risk related to the burning.

PAMS does not provide information about the measurement of the Ra situation or about the status of SFSP. However, measurements are mediated by standard means. The main problem regarding accessibility remains ensuring the electricity supply for PAMS in the case of SBO.

Within the project for the qualification of NPP Dukovany equipment for selected emergency events, the thermo-hydraulic and radiation parameters of the environment were stated. For each monitored area and for each monitored thermo-hydraulic parameter, the qualification envelope curve of the stated parameter was created from all known courses.

In the case of a severe accident, the conditions in the containment will be unfavorable in the long term at the level of harsh environment conditions. In particular, it is necessary to take into consideration the very high dose rate which may locally, due to the influence of aerosols, achieve extreme values and results and may damage the sensor. There are special conditions during the burning of hydrogen. For a few seconds, the temperature of the atmosphere may exceed 1000 °C and, according to the size of the fire, the pressure may exceed the design pressure of 150 kPa. Despite this, it is probable that the greater part of the measurement is also retained after the fire because the thermal capacity of the walls and equipment is significantly higher than the capacity of the atmosphere. The systems for Ra measurement in the containment (ISRC) and in the ventilation chimney (Kalina) are capable of continuing to work in the long-term in a harsh environment in the case of high radiation assumed during a severe accident. The other measuring equipment is less robust and resistant; a hydrogen fire could cause damage to the cable insulation. There is no complete information available on the behavior of impulse tubes and measurement of the level faces for a short time up to high temperatures during the burning of hydrogen.

All the mentioned influences could lead to a worsening of the precision of the measurement. In particular, it is necessary to consider the “unreliability” of data on temperatures. It is better to rely more on the data about the pressure which will be relatively balanced in the long term and the differences provide data, for example, on the function of the vacuum-bubbler system. The dose input power is measured by qualified instrumentation and the data can be used (with respect to the sedimentation of aerosols on detectors) directly. SAMG also rely on the measurement of doses outside the containment whose interpretation can be estimated as an release route.

Potential effects from the other neighboring installations at the site, including consideration of the restricted availability of trained staff to deal with multi-unit, extended accidents

No risks are identified for the influence of industrial equipment on the NPP Dukovany site. There is no such equipment in the surroundings of the power plant. The capacity of the shift personnel for starting the activity is sufficient, for long-term managing of the emergency statuses at the same time, then in all units the implementation of the personnel must be the subject of the special regime (relief and strengthening the personnel in exposed workplaces, rest, boarding and managing available sources).

II.6.1.4 Conclusion on the adequacy of organizational issues for accident management

The objective of managing (control) accidents in NPP is to ensure the 4th level of in-depth protection (decrease of consequences after the origination of an accident). This level relates to the emergency preparedness of the NPP as the 5th level of in-depth protection (decrease of consequences of accidents accompanied by releases of Ra substances).

NPP Dukovany implemented the system for managing accidents for ensuring the 4th level of in-depth protection and the system of emergency preparedness for ensuring the 5th level of in-depth protection. The functioning and connected system for managing accidents and accident preparedness is ensured in NPP Dukovany by the robust set of measures of personnel, administrative and technical character.

In the personnel area this concerns the existence of the emergency response organization and ensuring activities corresponding to individual functions; in the administrative area this concerns the implementation of the respective procedures, manuals and instructions with the use of the capacities of technical support centres; in the technical area, regarding ensuring the functionality of the requested scope of technical means for the implementation of strategies. Performing actions during the origination of an extraordinary event is ensured in the first (preventive) stage of the development of an event always by the personnel from the 24 hour shift operation. In the case that the event exceeds by its scope the framework of capabilities of the personnel from the 24 hour shift operation, the second phase will start (decrease of consequences) where the organization of the emergency response is activated. In this case the responsibility for the management of actions is on the part of the NPP Dukovany emergency board with the support of technical support centre.

In the case of the origination of an extraordinary event, all necessary activities must be managed and performed from shelters. TSC and ERB, which manage the strategies according to SAMG, are located in the ECC, which is a secured workplace with the option of occupation in the case of the release of radioactivity into the air. The remote activities for the implementation of the strategy would be carried out by personnel from CR or ECR, where the project is completed concerning the occupation of these control centres. The local activities and repairs to equipment in the respective parts of the reactor room, machinery room or external objects would be ensured by the action groups located in the operating supporting centre.

The concept of managing technological accidents at NPP Dukovany is based on symptomatic approach. For the solution of technological accidents, strategies were prepared which are contained in EOPs, whose main priority is the restoration of the heat removal from RC and prevention of damage to the 1st barrier against the release of fission products (covering the fuel). For the decrease of the consequences of accidents, strategies are prepared which are contained in SAMG, where the main priority is to prevent damage to the 3rd barrier (containment) , which at this time is the last unbroken barrier. EOPs and SAMG are regularly updated to include knowledge from exercises on the simulator or during emergency exercises and from other NPPs.

During a threat to safety on the unit in the site or in the case of the origination of a situation which cannot be managed by the shift personnel, the system of emergency preparedness is implemented. In the case of the announcement of some level of an extraordinary event (Alert, Site emergency, General emergency) the organization of the emergency response is activated which has an internal part (SOER), consisting of shift personnel and an emergency part consisting of experts from the NPP technical personnel who are on-call.

For selection of shift employees and for selection of employees for SOER, a system of requirements for qualification was introduced and further criteria taking into consideration their professional knowledge is taken into consideration. The preparedness of the shift and technical personnel for managing technological accidents is regularly verified during training on a full-scope simulator with the participation of TSC personnel during emergency exercises.

Management of extraordinary events (including severe accidents) is given in the internal emergency plan approved by SUJB.

After the origination of emergency conditions (design and beyond design basis events without damage to fuel) for fulfillment of EOPs requirements, all actual technical means are used within their design designation. SAMG consider for the required activities the use of all available systems and equipment or all available technical means even out of their design designation.

In the NPP Dukovany site there is an available LFRU unit which has the respective fire technology and is trained for action in any part of the site. The pumping technology of LFRU is one of the main mobile non-technological means that can be used to transport and pump the media.

The program for managing accidents at NPP Dukovany is analytically supported in the long term. The analytical support is based on the probability – deterministic approach which is based on the selection of the most probable emergency scenarios leading to severe accidents and, consequently, their deterministic by means of calculating codes. The result of the analytical support is the knowledge based on the understanding of phenomena during severe accidents and their timing, identification of possible weaknesses in the design, determination of the activity for decreasing the consequences of severe accidents, validation of activities for the response to severe accidents and the determination of source members for the evaluation of any radiological consequences. At present the simulation tool is completed for the display of phenomena during specific scenarios of severe accidents.

II.6.1.5 Measures that can be envisaged to enhance accident management capabilities

Even though there are several diversion systems for the implementation of each strategy for managing accidents, in the areas of abilities to manage severe accidents opportunities to further increase safety were identified.

In the area of the administrative solution, this concerns manuals for managing severe accidents for shutdown statuses (SAMG for shutdown statuses), which are not ready in NPP Dukovany. Nevertheless, for the maintenance of EOPs, SDEOPs and SAMG, the “Maintenance program” is regularly implemented or updated.

In the personnel area, problems can occur with the accessibility of the site or the usability of ECC and the managing of activities and decisions about high risk versions of the solution when managing an emergency situation and with the communication and warning of the personnel.

The plans for further considering the existing system aimed at evaluating the preparedness for management of extraordinary situations from backup emergency centres (in the case of inaccessibility of the site) and in the periodical investigation of the nomination of the most professional SOER personnel.

To increase the efficiency of the system for managing accidents, measures in the following area will be prepared:

1. The organizational ensuring of the most effective use of existing capacities or definition of additional capacities – the crisis plans for managing the predictable statuses of the NPP (affecting the whole site, loss of control centres for emergency preparedness, loss of notification and warning systems, decision on high risk versions of the solution, shift rotation of personnel, extreme natural conditions , …).
2. Completion of technological regulations (procedures) manuals for managing the selected beyond design and severe accidents (SAMG for shutdown units, SAMG for damaged fuel in SFSP, EDMG, …) with the aim of ensuring the cooling and heat removal from RC and SFSP and prevention of radioactive releases.
3. Increase in the level of training of personnel in the area of managing severe accidents (use of simulation tool for displaying the courses of parameters, phenomena and behavior of the unit during specific scenarios of severe accidents).
4. Additional technical measures for ensuring non-technological support functions (access to objects, access to fire technology, enduring ECC and safe places, physical protection system, …).

5. The alternative means for ensuring the long-term functionality of the communication between all elements of the system for managing accidents.

Opportunities for improving defense in depth are mentioned in the following table.

Some of the measures (in the note indicated “Finding PSR”) would also be implemented without this target oriented evaluation which by its outputs confirms the efficiency and correctness of the previously accepted decisions for the implementation of measures to make the original design more resistant.

Tab. 21: Opportunities for improving defense in depth to enhance accident management capabilities

Opportunity for improvement	Corrective measure	Deadline (short-term I / medium-term II)	Note
Staffing of CR during severe accident	Ensuring staffing of DB	II	PSR finding
Occupation of safe places during severe accident	Oxygen regeneration in safe places	II	PSR finding
PAMS	Completion of measurements of the Ra situation and the status of SFSP	II	
Regulations	Prepare “shutdown SAMG” for shutdown / Severe accident in SFSP	I	PSR finding
Emergency preparedness	Ensuring alternative means for warning and notification of NPP Dukovany personnel and inhabitants in EPZ	I	
Regulations	EDMG manuals for the use of alternative means	II	
Emergency preparedness	Ensuring the functioning of emergency response elements in the case of non-accessibility of ECC	I	
Personnel	Introduction of TSC training in the area of severe accidents	I	Finding PSR – Implementation of simulating tool for displaying severe accidents

Emergency preparedness	Preparation of agreements with external elements (IRS, army) and nearby NPP. Organizational measures	II	

II.6.2 Accident management measures in place at the various stages of the scenario for loss of the core cooling function

II.6.2.1 Prior to fuel damage in the reactor pressure vessel / a number of pressure tubes (including the last resort for preventing fuel damage)

The basic reason for severe accidents is the insufficient collection of residual heat released from the fuel in the RC. Damage to RC is considered when the temperature of the coverage locally exceeds 1200 °C, when a steam-zircon reaction develops. Due to the impossibility to measure this parameter, the setpoint for transfer into SAMG was set for values on the output RC 550 °C. Exceeding of 1200 °C in the wider area leads to an intensive steam-zircon reaction which is exothermic. A greater volume of heat is released than the residual heat; this heat contributes to the development of the accident because it is mostly accumulated inside RC.

The restoration of the heat removal from RC on the part of II.C by alternative means is performed in EOPs i.e. before the transfer to SAMG. In addition, activities are performed related to the de-pressurizing of I.C with the aim to enable injection of low-pressure pumps into I.C.

There are two ways of stopping the development of the loss (cooling of RC up to a severe accident):

- The restoration of the heat removal through SG (alternative filling of SG with low-pressure sources, including adding water by means of LFRU).
- Heat removal by adding coolant into I.C and discharging from the exit hole in the primary system (at LOCA) or by open pressurizer valves (feed&bleed).

EOPs also include alternative strategies:

- De-pressurizing of the primary system or cooling on the part of II.C, which may lead to the enforcement of a hydraulic accumulator or, even, low-pressure emergency or alternative sources.
- The restoration of the operational ability of high-pressure systems for emergency filling or alternative high-pressure systems for emergency filling I.C.
- Use of the remaining coolant in loops enforced by the start of MCP even at the cost of its destruction.

II.6.2.2 After the occurrence of fuel damage in the reactor pressure vessel

Conservatively, it is possible to reconnect damaged places at the moment of the beginning of the steam-zircon reaction related to the massive production of hydrogen which preceded the start of the loss of RC geometry. The symptom of damage of RC by melting is, in addition to the increased temperature, mainly an increase in the concentration of hydrogen in the containment. Due to the speed of the production of hydrogen against the loss of geometry, the concentration of hydrogen would not be managed fast enough by existing re-combiners. However, there is a time reserve (several tens of minutes) for the safe burning of the hydrogen in the initial phase.

The typical period from entrance into SAMG into breaking the integrity of RPV by affection of the melted reactor core RPV is about 7 hours if all forms of delivery of the coolant into the vessel failed.

The basic strategy that is used in SAMG in this phase of the accident is to decrease the pressure in the primary system either due to the decrease in the production of hydrogen, as well as the prevention of creep breaking of the bottom and high pressure expulsion of the melt from the vessel. In accordance with generic manuals, the value of pressure I.C is required to be under the value 2 MPa, in particular the value 1 MPa is stated for NPP Dukovany units. The phenomenon called “direct heating of the containment” at VVER-440 in the whole volume is not a risk and is improbable. However, the loading of the wall of the reactor shaft with high overpressure of gases from the primary system is risky, if the hole is sufficiently large. This loading can be reinforced by the “direct heating” of the atmosphere in the shaft from the particles of the melted fuel after the failure of the vessel.

The decrease in the pressure in the primary system is one of the highest priorities due to the prevention of expelling debris from the vessel under high pressure. The damage to the fuel by the steam-zircon reaction under high pressure significantly increases the production of hydrogen; therefore, the decrease of pressure is necessary before the risk of failure of the bottom of the vessel. For this purpose, it is possible to use PSV and PRV.

Between the start of the damage to RC and the failure of the vessel, it is possible to distinguish several partial phases corresponding to the gradual loss of geometry, creation of the pool of the melt on the load-bearing board and its failure and the dropping of debris to the bottom of the vessel. Due to the large volume of water in the lower part of the vessel, the failure of the bottom occurs after several hours, and the existence of water cannot prevent it because the accumulating debris creates a configuration that cannot be cooled.

The strategy of the restoration of the heat removal is resolved in SAMG by de-pressurizing and filling I.C. In this phase of the accident it is not possible to use the cooling I.C from the side II.C, and, therefore, it is necessary to add the coolant directly into the reactor vessel.

The earlier the delivery of water, the greater chance there is to stop the accident and keep the melt in the vessel. With the exception of the case where most debris is on the bottom of the vessel in an unfavorable configuration, there is a definite chance of preventing the failure of the RPV bottom. Therefore, SAMG manuals recommend starting the delivery of water in the instance when it is managed to restore the source in a volume greater than the minimum flow necessary for flooding the RC. It was stated in the SAMG as such a flow that will be evaporated by the residual heat of the RC.

The risk of the failure of the vessel would be significantly decreased by the implementation of the strategy for cooling the vessel from the outside by flooding the reactor shaft. The success of this strategy was analytically confirmed. Within the implementation of the technical solution for the modification of piping lines for the air ventilation system into the reactor shaft, glow holes were prepared from the floor of the SG box enabling the termination of the flooding of the room in the reactor shaft. The level was previously measured in the room in the reactor shaft and after blinding the drainage into a special sewerage system. Both these events support the above-mentioned strategy.

II.6.2.3 After failure of the reactor pressure vessel

If the accident cannot be stopped inside the reactor pressure vessel, there would be the failure of its lower part and the interaction of the melted fuel with the concrete. The main consequences of this phase of the accident could be as follows:

- additional production of hydrogen from non-oxidized Zr, steel in the debris and armoring of the concrete.
- penetration of the melt through the wall of the reactor shaft.

Production of hydrogen during the interaction of the melt with the concrete at the shaft bottom of the reactor is not as fast as during the oxidation of the covering. This concerns two orders lower than the production of hydrogen during the reaction of the water steam with the zircon coverage.

The penetration of the melt through the wall of the shaft is more serious than penetration through the bottom of the shaft because:

- the penetration of the melt in a radial (horizontal) direction is faster than penetration in the axial (vertical) direction.
- the wall is 2.5 m thinner than the bottom 3.1 m.
- the wall of the shaft represents the border of the containment; debris can penetrate through the bottom into the foundation slab (bed) where fissile products are kept.

The damage to the bottom would be caused by thermal creep. In the case of higher pressure inside the vessel, it can be at the lower temperature of the bottom and before the melting of the debris on the bottom. The location of the damage depends on the pressure; in the case of higher pressure, it is in the place of the maximum stress on the bottom of the vessel; in the case of lower pressure it is in the place of the thinning of the vessel in the cylindrical part.

After damaging the RPV vessel, there could be the relocation of materials from the vessel and a layer of debris would be gradually created with a dense arrangement that could not be cooled because there would be melting of the debris and the interaction with the concrete under the layer of water which is insulated by a boiling film. During passing the debris through concrete there would be forming of CO, steam and hydrogen.

On the surface of the melt in the shaft, a firm crust would be created which would restrict the heat removal from the surface of the melt and also in the case that it would be covered with water.

For reactor VVER-440 the door in the shaft would be protected for a certain period against contact with liquid debris by solid debris or the shell. However, the crust has low thermal conductance. Due to the foaming of the melt and the low temperature of the steel melt, most probable is that there would be melting of the lower part of the door and flowing of part of the melt through a lane to the second door which would fail after some time. In each case it is not possible to exclude minor damage of the containment shortly after the failure of the bottom of the vessel due to the failure of the rubber sealing on the door.

The door could be protected by flooding the debris in the shaft. Even in the case of breaking the sealing, there is reserve sealing for the outside door that could prevent the release of water and protect the door. This method of protection of the door was not analyzed; everything is based on a professional estimate.

If measures for preventing the failure of the door are taken, then there could be penetration of the melt through the wall of the shaft after about 4,5 days after the failure of the vessel bottom. This represents high and late damage to the containment. The concentration of fission products in the atmosphere of the containment would be low at this time.

The strategy for cooling the melt is part of [SAG-5] for flooding of the shaft. The existing configuration of the power plant provides the possibility to flood the shaft by pouring; for this, it is necessary to have water from two TH tanks and bubbler condenser water trays. Therefore, the manual considers discharging the bubbler water trays, including the check of the closing of the drenage of the SG box. For pumping water from the TH tanks, it would be possible to use spray pumps TQ, alternatively TG and TM system pumps. The strategy also considers the use of the water reserve from the surrounding units.

The main objective of the strategy of flooding debris in the reactor shaft is the cooling of the steel door and confinement the fission products released during the interaction of the melt with the concrete.

II.6.3 Maintaining containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core

II.6.3.1 Elimination of fuel damage / meltdown at high pressure

Design provisions

The basic design tool for ensuring the depressurizing of I.C is PSV. A further option is to use PRV (where the flow section is lower than for PSV) and re-pressurizing SG.

In EOPs there are strategies for keeping the long-term safety status based on the managed cooling and de-pressurizing of I.C. One of the main priorities of SAMG is to prevent damage to RPV by melting in RC with high pressure; activities are also started in EOPs due to the risk of the increase in the production of hydrogen and consequently after entrance into SAMG.

Operational provisions

Even if there is no de-pressurizing of I.C before the massive steam-zircon reaction, there is sufficient time for the prevention of the failure of the vessel under high pressure. Phenomena related to the production of hydrogen, such as burning or explosion, may influence relations inside the containment and also the control of PSV and PRV.

II.6.3.2 Management of hydrogen risks inside the containment

Design provisions, including consideration of adequacy in terms of the hydrogen production rate and amount

The integrity of the containment in the early phase of the accident is mostly threatened by a large fire or detonation of hydrogen following the failure of the double door in the reactor shaft. In the late phase, there is the penetration of debris through the shaft. The threat to the containment by hydrogen could take place after the damage to RC during the steam-zircon reaction. With the influence of the large surface of the coverage and the exothermic reaction, the development of hydrogen is very fast, between 0.5 and 1 kg/s. Due to the speed of the production of the hydrogen before the loss of geometry, such a volume of hydrogen cannot be managed by the existing re-combiners. The development of the hydrogen would also continue in the late phase of the accident during the reaction of the melt with the concrete on the bottom of the reactor shaft; however with two orders lower speed (less than 0.01 kg/s). In terms of the threat to the integrity of the containment by the hydrogen in the late phase, the risk should increase provided that that up to this time the containment remained integral. It is improbable that there could be a burning large volume of hydrogen in the early phase; in the worse case there could be fast burning or detonation that would lead to irrevocable damage to the containment and the hydrogen would release. In the case of RC debris expelled from the reactor vessel, they are also important flaming sources of hydrogen.

The kinetic production of the hydrogen takes place about 30 minutes after exceeding the temperature of gas at the output RC 550 °C. The course of the production of the hydrogen from the steam-zircon reaction is significantly intensive at high pressure, so one of the primary requirements of SAMG is the instruction for de-pressurizing I.C. The greatest part of the hydrogen is produced during the period of one hour before relocation of the melted fuel.

The hydrogen released into the containment could lead to dangerous explosive concentrations in the bubbler condenser shaft where there is a lower concentration of the water steam due to the condensing effect on the channel surfaces. The strength of the containment could be threatened in the case of burning of hydrogen in large volume in the whole containment.

The main sources of hydrogen during a severe accident are from the steam-zircon reaction on the surface of the clading and the shells of the fuel in the reactor or reactions of steam released from the concrete in the reactor shaft with metals in the debris.

Containments of NPP Dukovany units are fitted with a system for liquidating post accident hydrogen which is solely designed for design basis accidents. For the LOCA accidents where only a very low volume of hydrogen is produced, then for liquidation there are 17 combiners available in the containment. The increase in the robustness of the NPP Dukovany for severe accidents was decided after the Periodic Safety Review in 2006. In the final phase of the preparation, there is a project for the construction of the system for effective liquidation of hydrogen which will be able to manage hypothetically originated hydrogen in the case of the worst scenario (in terms of the production of hydrogen) in a severe accident. The current analyses and experience from other VVER confirmed that such a system consisting of powerfull re-combiners (approx 30 pieces) completed with burners in the case of functioning spray system, can restrict the risk of the flame spreading and exclude the risk of the transfer to detonation.

Operational provisions

The threat to the integrity of the containment by burning the hydrogen is resolved by SAMG either on the principle of intentional burning or inertization of the containment. Each accidental or intentional burning of the hydrogen or its re-combination decreases the concentration of oxygen in the atmosphere and restricts the future risk by burning the hydrogen. For the full consumption of oxygen in the part of the containment without air traps, it is sufficient to burn re-combinations of about 700 kg of hydrogen. Further hydrogen produced during the interaction with the concrete only increases the pressure in the containment and does not contribute to the risk of burning the hydrogen (because there is no oxygen).

The ignition is to create a spark by using the electrical equipment inside the containment. The manuals contain the list of equipment with which the CR personnel tries to manipulate (change the position of the valves) the air so as to initiate sparks. When using equipment at lower height levels, burning would be easier. The use of electrical equipment is not always completely effective. Most of these devices are in the insulated non-sparking version. However, there is a short time for ignition; in the case of quick production of hydrogen, the contraction of the hydrogen will be too high.

For inertization of the containment it is possible to discharge the nitrogen from the hydraulic accumulator; for effective inertization in the existing status of the design, it is possible to use water steam which postpones the risk of burning at a higher concentration of the hydrogen. However, with the high probability there will be the burning of the hydrogen by the existing re-combiners as long as its concentration does not exceed 10% at the point of installation. The existing re-combiners do not resolve the risk of the hydrogen during a severe accident because they can remove only several kg of the hydrogen in the early phase of the accident.

In the case of de-pressurizing I.C before the damage to RC (which is performed within EOPs) and the continuation of this procedure after the damage to RC, the risk of detonation is later and localized only in the bubbler shaft.

II.6.3.3 Prevention of containment overpressure

Design provisions, including the means to restrict radioactive releases if prevention of overpressure requires steam / gas relief from containment

The purpose of the design function of the containment is to prevent the release of Ra substances into the environment, or to restrict the radiation consequences of the accident in the surroundings. The containment represents the last barrier against release and is independent of the other barriers. The function of the containment is ensured by the construction and the structure which definitely resists design over-pressure of 150 kPa and with most probability, double over-pressure. The tightness of the containment is regularly inspected (within the PERIZ tightness test) and measures are taken that increase the tightness.

The projected function of the containment is ensured in two ways:

1. With the use of insulating FAV in all routes passing through the wall of the containment, use of sealed passages and sealed penetrations of all pipes and cables passing through the wall.
2. Minimizing of releases by restricting the duration of inside overpressure with consequent creation of under-pressure towards the surroundings.

The system for the suppression of pressure in the containment consists of two parts:

- Vacuum-bubbler condenser system containing passive functioning water trays which condensate water steam and consequently ensures passive spraying of the containment. Non condensed gases and air from the containment area are located in the air traps which are consequently automatically separated from the environment of the containment.
- Spray system with three active spray pumps.

The cooperation of both systems creates under-pressure in the containment and full elimination of the release into the surroundings. The correct activity of the bubbler condenser which is essential for the fulfillment of the function of the VVER 440/213 containment, was verified within the Project PHARE/TACIS PH2.13/95 “Experimental qualification of the bubbler condenser”. Tests, experiments on unique equipment which models SG and VBC boxes in the scale 1:100 and final analyses showed that the vacuum-bubbler system for nuclear power plants VVER 440/213 (Paks, Dukovany, Bohunice and Rovno) are able to resist the loads and to maintain functionality. It is the principal restriction which limits the maximum pressure during accidents with major releases. This ensures the maximum contribution for the reduction of pressure to the under-pressure immediately after the start of the LOCA accident with large releases and prevents the release of radioactive materials into the environment. During the development of a severe accident it is not possible to permanently keep the under-pressure in the containment, it implies from the results of the analyses that it is possible to guarantee the minimum overpressure and the release of the radioactivity will be lower than 0.10 of volatile fissile products with the exception of rare gases. In the case of hypothetical failure of active sprays, the bubbler system ensures a lower pressure in the containment than in the case of traditional full-pressure containment and the release into the surroundings is less than 1% of volatile fissile products with the exception of rare gases. The vacuum-bubbler system eliminates the lower tightness of its containment compared to full-pressure containment.

This is valid for the containment while keeping its integrity. After its loss it is necessary to consider the very high release of the radioactivity into the surroundings which could be partially restricted by the active spray system.

The threat to the NPP Dukovany VVER-440/213 containment by the over-pressure of gases (with the exception of a short increase in pressure during burning of the hydrogen) is very small. It relates to the following rights:

- The vacuum-bubbler system condenses the steam and creates at the start of the accident the conditions for under-pressure in the containment at the costs of certain pressurizing of its parts – air traps.
- The total volume of the containment, including the air traps, compared with the residual power, is relatively high, approximately 50,000 m^3.
- The relatively high operating leakage of the containment of a few percent of weight of gas/day at the design pressure supports the decrease of the pressure. As the leakage

has the character of small cracks in the concrete, due to the influence of aerosols, including the water fog, these ceacks may be blocked.

The 250 kPa pressure (overpressure 150 kPa) is the design pressure where major damage to the containment is improbable. According to the strength calculations, for NPP at the over-pressure of about 290 kPa there is a risk to the integrity of the containment with about 5% probability, over-pressure of 350 kPa corresponds to 50% probability.

The results of the analyses of the possibility of the loss of integrity of the containment by over-pressure of the hydrogen shows that after 4 or 5 days, at the time of the penetration of debris through the wall of the shaft, the over-pressure in the containment would be about 120 kPa; according to the estimate, if there is no failure of the shaft wall then it would achieve the projected over-pressure after about 5 days. Blocking the leakage would have a major influence on the course of the pressure, if it does not take place, the maximum over-pressure after 4 or 5 days would be about 60 kPa.

The pressurizing of the containment by water steam in the case of the failure of the heat removal and the spray system may be theoretically faster than pressurizing by hydrogen. However, this scenario can be excluded in practice because in the case of the failure of the heat removal, there is probably the loss of water and the interruption of the production of steam.

Operational and organizational provisions

The strategy for the prevention of over-pressurizing is described in SAMG, Control of pressure in the box which is used at the over-pressure 10 kPa. Its meaning is it is better to prevent the higher release of existing leakages than future threats to the containment by over-pressure. This corresponds to the use firstly of the systems for containment with the heat removal as the spray system or recirculation ventilation while keeping the pressure border of the containment.

The manual SAMG "Decrease of the pressure in the box" considers, in addition to these systems, the use of air ventilation systems for the controlled **venting, use of systems** equipped with filters or aerosols and iodine **(however, they are not designated for this purpose)**. In this case, the release of radioactivity would be restricted because part of the rare gases would remain caught in the bubler condenser air traps.

It can be stated that the strategies for the prevention of overpressure of the containment can be based on the existing means as it was analytically confirmed that it is not necessary to develop a further special filtered ventilation system for severe accidents.

II.6.3.4 Prevention of re-criticality

Design provisions

For VVER-440 the risk of the solution of boron in the advance phase of the accident is lower than for the PWR reactors. Due to the insertion of control rods with pulling out the part of the fuel assemblies (37 of 349) from RC the reactivity would also be lower in the case of melting and relocation of control rods.

Operational provisions

In the status of the threat of under-criticality of RC in the preventive phase (EOPs), the increase of the concentration of boron is required due to the non-insertion of E&CR and not to compensate for contributions to the positive reactivity from the decrease of the temperature during cooling.

After the loss of the geometry of the fuel, the problem of the boron solution does not exist. The geometry created by debris inside the reactor or in the shaft under the reactor is under all situations deeply under-critical even in the case of a clear water flow.

II.6.3.5 Prevention of basemat melt through

Potential design arrangements for retention of the corium in the pressure vessel

Design of VVER-440/213 is suitable for holding the melt inside the vessel by cooling the vessel from the outside, although the original design did not consider this measure. In particular, the residual heat of the reactor is very low which ensures low thermal flows on the outside surface of the vessel in the area of bubbler boiling with a large reserve to the boiling crisis. The vessel does not have any penetrations in the lower part. The reactor shaft is the lowest place in the containment and in the case of the loss of emergency water discharging of the bubbler water trays is sufficient for its flooding.

The increase of the robustness of the NPP Dukovany for severe accidents was decided after the Periodic Safety Review in 2006. Some modifications were made to the NPP Dukovany units aimed at cooling the vessel from the outside. In particular, the sewerage system on the bottom of the shaft is closed, measurement of the level in the room in the reactor shaft is completed and the supply line for ventilation TL11 into the reactor shaft room is modified, including the preparation of the inflowing holes so that it can be fitted with charging valves. It is necessary to make certain modifications in the lower part of the vessel so as not to prevent the access of water to the vessel and small modifications in the lower part of the shaft room (screens) and in the upper part of the room (collection of steam into the containment from the area of the reactor shaft).

Potential arrangements to cool the corium inside the containment after reactor pressure vessel rupture

A highly effective measure for protection of the containment before the late phase of the accident (and related problems to the restoration of the source of hydrogen, melting of the door or the shaft) would be to keep the melt inside the vessel by flooding the reactor shaft.

The consequences of the melting of the door in the shaft can be decreased by hermetization of room A0065 or by sealing the semi space between the steel doors with concrete units. The cooling of debris in the shaft with water can be effective.

The interaction of the melt with the concrete at the bottom of the containment is for VVER-440/213 a well known phenomenon related to the penetration of the melt through the wall of the shaft or prior to this, through the double steel door in the wall of the shaft into non-hermetic premises.

The slower phenomenon is the melting of the foundation slab. It is thicker than the wall of the shaft, 3.1 m, and underneath is earth which would contribute to the filtering of fission products. The procedure for penetration of the melt is faster in a radial than in an axial direction.

The development of the hydrogen in this phase of the accident compared with the development of the hydrogen during the steam-zircon reaction in the early phase of the accident is significantly slower.

The side penetration of debris through the wall of the shaft, according to the analyses, would take place about after 4 or 5 days from the initiating event provided that before this there was no melting of either steel door. Water delivered into the shaft after the penetration of debris through the vessel could extend this time and it could protect the steel door.

The strategy for cooling the melt is part of SAMG, "Flooding of the reactor shaft". The delivery of water into the shaft before the melting of the vessel is only possible at present by pouring through the reactor footstall. A large volume of water is necessary for it, the reserve of both TH tanks and the water from the bubbler channels. This intervention may prolong the time before the penetration of debris through the shaft by several hours. The manual proposes the discharging of the bubbler channels or pumping of the water from TH tanks provided that the TQ pumps are operable. The strategy also considers the use of the water reserve from the surrounding units.

After termination of the technical solution, the cooling of debris in the shaft would be redundant because debris would be caught in the vessel. However, irrespective of this fact, there is the breaching of the bottom of the vessel, the shaft would be flooded and the steel door would be automatically protected and the interaction of debris with the concrete would be postponed.

The main contribution of the strategy for the flooding of debris in the shaft is the cooling of the steel door. In the case of a large volume of debris, the water will not completely stop the penetration of the melt through the concrete in the shaft.

Cliff edge effects related to the time delay between reactor shutdown and core meltdown

It results from the analysis of the scenario that when there is a loss of the heat removal from I.C on the part of SG, even without performing the alternative activities described in EOPs, there is a relatively long time reserve for restoration of the heat removal from I.C. The temperature on the output of RC 550 °C was reached in about 9 hours from the origination of SBO in the case of not performing the activities required in the preventive phase within EOPs. The similar time reserves were also ascertained during the “transient” scenario (full loss of SG supply). In the case of alternatively filling the SG in accordance with EOPs it is possible to efficiently prolong this period by up to several days.

The LOCA accident with the loss of all active systems for the emergency filling of the primary coolant could theoretically lead to the damage of RC. An example of such an accident is the combination of SBO+LOCA. However, the results of PSA show extremely low frequencies of these events, less than 10^{-8}/year. The analyses of severe accidents are aimed at the most probable LOCA scenarios, where loss of cooling occurs in the recirculation phase of the operation of ECCS emergency pumps (transfer to the suction from the containment). In most cases it is possible to consider damage of RC by discharging the bubbler channels so the damage of RC would take place later than in the case of SBO with the failure of the alternative methods of filling SG.

II.6.3.6 Requirement and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

Design provisions

The integrity of the containment is ensured by the following systems:

- System of insulation of the containment – FAV controlled by compressed air (they have compressed tanks) and FAV controlled electrically on ventilation systems (the operational capability is conditioned by the supply of the SSPS I. category)
- The system of liquidation of the design basis accidents event contains passive auto-catalyst re-combiners (also after possible completion is more powerful) and does not require an electricity supply
- The passive part of the vacuum-bubbler system contains mechanical valves which do not require an electricity supply
- The active part of the vacuum-bubbler system – active spray system requiring SSPS II. category – and its long-term failure does not lead to the loss of integrity of the containment.

Operational provisions

Most strategies described in SAMG (adding water into the containment, heat removal, keeping pressure in the containment) require for successful implementation at least restricted accessibility to the electricity supply.

II.6.3.7 Measuring and control instrumentation required for protecting containment integrity

NPP Dukovany units are equipped with a system for measuring the concentration of hydrogen. The system for the concentration of the hydrogen contains 15 sensors with a range up to 10 % located in various rooms of the containment.

After a severe accident there can be certain damage to some measuring sensors by fire from hydrogen. Despite this, most measurements necessary for performing SAMG should be accessible.

The limit for the measurement of the concentration of 10 % hydrogen would be exceeded in the case of a hydrogen fire. The restricted range of measurement of the concentration of the hydrogen does not represent too great a problem because the transfer from the strategy based on the ignition of the hydrogen to the strategy for the prevention of burning would be performed at lower concentrations of hydrogen than 10 %. The problem related to managing the issue is the absence of the measurement of the concentration of water steam in the containment. Therefore, SAMG contains calculating aids which replace the missing information.

For the evaluation of the loss of cooling of RC, the measurement of temperature on the RC output is used. It is necessary to consider the gradual loss of thermo-cell on the RC output. For confirmation of the efficiency of individual strategies, the measurements of levels and measurements of pressure are important. In addition to the measurement of the level in the shaft, they will probably be accessible and undamaged. Their part is outside the affected containment – TH tanks and all measurements in the neighboring unit.

The measurement of the pressure and the water level on the floor of the containment that function during the normal operation of the unit and during emergency statuses were not especially designed for severe accidents. However, their partial retaining in the case of a severe accident is proposed. Decisive for the protection of the integrity of the containment is the measurement of pressure on the basis of which the decision to perform the filtered and not filtered ventilation of the containment is issued for the protection of the integrity during the growth of pressure above 300 kPa (if such over-pressure is achieved). The measurement, as well as the respective action, requires the functioning of SSPS I.

For the evaluation of the threat of the containment by the over-pressure, there are measurements available for the over-pressure in the containment with the sufficient range (450 kPa), which are located in other places. It is expected that at least one of the measurements should resist the course of a severe accident.

For the identification of the interaction of the melt with the concrete at the bottom of the containment there are no direct measurements, although it is possible to identify it from the measurements described below and to estimate according to the calculating aid mentioned in SAMG, however the mentioned indicators are not as exact as direct measurements. The interaction of the melt with the concrete at the bottom of the containment can be identified on the basis of indirect measurements which are accessible in PAMS. These measurements are measurements of the pressure and level in SG boxes and in the shaft and the reactor vessel if this previous measurement resists the conditions of a severe accident. The indication of the pressure in the primary system is better which decreases the pressure in the containment (if it did not decrease before). Starting a new increase in the concentration of the hydrogen in the containment could be a further indicator, the content of hydrogen in the containment would probably be outside the measuring range of 10 % or highly inexact due to the influence of depletion by oxygen.

The basic measurement in the case of large releases of fission products are measurements of dose rates and radioactivity. For the measurement of dose rates and radioactivity, it is possible to use the measurement in the containment outside the containment, dose rates and the activity in chimneys and measurements from the tele-dosimetric system located on the fence of the power plant. The ranges of all these measurements are proposed for operation, as well as the emergency and post-emergency conditions.

The radiation measurements with ranges for emergency and post-emergency conditions are implemented in CRMS system and are not implemented into PAMS yet.

II.6.3.8 Capability for severe accident management in the case of simultaneous core melt/fuel damage accidents at different units on the same site

NPP Dukovany units, in terms of construction, are linked into twinss, however, technologically, the individual units are mutually independent. Moreover, the twins arrangement enables, in the case of an accident, also the use of media from the neighboring unit using common auxiliary systems. Activities when managing accidents on individual NPP units are managed from ECC (TSC and ERB) and actions on individual units are carried out by the operating personnel of the respective unit. According to the actual situation in the individual units, it is possible to operatively transfer capacities from one unit to another. During the origination of an accident on one unit, TSC personnel have instructions to take decisions on the manner of operation and carrying out the necessary activities on the neighboring unit. In the case of an event developing into a severe accident on further units, the same SAMG manuals for all units would be used; nevertheless the situation on individual units would be evaluated independently and TSC and ERB would implement the necessary coordination between the activities on individual units.

II.6.3.9 Conclusion on the adequacy of severe accident management systems for protection of containment integrity

For managing the above-project and severe accidents, all available hardware is used, including that which is not priority projected for managing severe accidents. The use of these means is described in the respective strategies contained in EOPs and SAMG. Strategies are oriented towards success, i.e. one of the side objectives of SAMG is the restoration of the operational capability of systems and equipment within the maximum range and the implementation of the stated strategy in any described manner leading to success. Success means the fulfillment of main SAMG objectives, i.e. putting the unit into a stable status that can be controlled by restricting the release of radioactive products.

Even if there are several diverse systems for the implementation of each strategy, within the evaluation of measures for the protection of integrity, the possibility of the release of radioactive products into the surroundings was identified due to the threat to the integrity of the containment by hydrogen during a severe accident and restricted possibilities for the prevention of the loss of integrity of the containment due to the melting of RPV and consequently, the bottom of the containment.

Some measures for managing accidents may be very risky (threat to per SÚJB large releases of radioactivity, destruction of NPP, etc.). These risks must be carefully evaluated in advance and the employees responsible for managing the emergency situations must be able to decide whether to take such high risk measures.

The long-term post-emergency activities in terms of managing severe accidents are based on the continuation of activities after ensuring the heat removal and eliminating the occurrence of high-energy phenomena of the burning or explosion of hydrogen, etc. depending on the status of the unit. In this case, it is highly problematic to exactly define in which status the unit is and to define any threats. Nevertheless, after putting the unit into a stable status that can be controlled, the basic precondition for the termination of SAMG is fulfilled. Before leaving SAMG and continuing with long-term post-emergency activities, it is described within SAMG in which manner it is possible to identify the status of the unit as precisely as possible to state the scope of the damage and long-term risks.

The long-term post-emergency activities are moved from the phase of searching for a suitable measure into the phase for ensuring the long-term functionality of the ascertained and applied successful measures, i.e. ensuring that there is no failure of the alternative sources of delivery of water for any reason (loss of supply, exhausting the reserve of water, failure of the components). Therefore, it relates to searching for alternatives to already successfully implemented measures, i.e. searching for further measures which, after ensuring, immediately replace already implemented measures or are in reserve in the case of the loss of actually implemented measures.

II.6.3.10 Measures that can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

For the increase of the capability to keep the integrity of the containment after serious damage to the fuel, means were proposed within PSR for ensuring the integrity of the containment (i.e. prevention of the release of fission products) during a severe accident (liquidation of hydrogen, localization of melt).
The opportunities for improvement of the in-depth protection during events where the consequence can be the origination of a severe accident are mentioned in the following table. All of the measures (in the note indicated “Finding PSR”) would also be implemented without this target oriented evaluation which by its outputs confirms the efficiency and correctness of the previously accepted decisions for the implementation of measures to make the original project more resistant.

Opportunity for improvement	Corrective measure	Deadline (short-term I / medium-term II)	Note
Integrity of the containment during a severe accident.	Increase of the capacity of the system for the liquidation of emergency hydrogen.	II	PSR finding
Localization of reactor core melt	Cooling of the melt from the outside of RPV	II	PSR finding

II.6.4 Accident management measures to restrict radioactive releases

II.6.4.1 Radioactive releases after loss of containment integrity

Design provisions

Uncontrolled releases of fission products from the power plant after damage to RC may represent a threat to the health and safety of the inhabitants. Large release (General emergency according to the internal emergency plan) is defined as an release which exceeds the criterion for the announcement of radiation EE-3 according to the Emergency plan.
Depending on the course of the accident, the activity can be released:

- directly into the containment and after integrity into the surroundings.
- dthrough SG into the secondary system and the surroundings.
- into a non-hermetic room.
- into the ESW system.

During the evaluation of fission products, the biological effects of ionizing radiation were taken into consideration which has a double character – stochastic and deterministic. In terms of the stochastic effects during which there is a limit value, it is not possible to state any marginal conditions. In terms of deterministic effects, when stating the marginal conditions, it is necessary to base them on the respective legislation and the normative documentation. The stating of limit conditions in terms of the biological effects of fission products is outside the framework of this evaluation.

Operational provisions

Strategies are described in the SAMG, manual "Restriction of the release of fission products" and the "Decrease in the consequences of the release of fission products". Individual activities for the restriction of the release of fission products differ according to the route through which the fission products are released. In the case of release into the containment and damaged or untightened containment into the surroundings, the primary method is to decrease the release of the TQ spraying system which is highly effective in the case of inaccessibility of ventilation systems equipped for the heat removal.

A primary and highly effective measure during the release through SG is to close the HUA equipment on the damaged loop, the further measures mentioned in SAMG can be implemented on the SG steam lines.

In terms of personnel and inhabitants, within SOER a fast mobile monitoring group was established which monitors and evaluates the radiation situation on the affected sectors. For the requirements of the preventive measure announced for the protection of inhabitants, there is SW RTARC.

II.6.4.2 Accident management after uncovering the top of the fuel in the fuel pool

Pools for the storage of fuel are located in the reactor hall which is common for both units. An analysis during accidents in the storage pool for shutdown statuses is planned for 2012. The behavior of the pool in mode 6 will be analyzed, i.e. replacement of the fuel, in mode 7 during the complete removing of the fuel from the reactor and in modes 1 to 5 when the pool for the store and the reactor hall are hermetically separated from the containment.

The storage pool for spent fuel (SFSP) is cooled by two cooling circuits. Each cooling circuit contains a circulating pump and heat exchanger. Heat exchangers are cooled by technical essential water (ESW 1 and ESW 3). SFSP cooling pumps and ESW pumps are also supplied from DG.

For the alternative filling and collection of the heat from SFSP in the case of the complete loss of normal SFSP cooling (either due to a decrease in the level, loss of coolant and interruption to the heat removal) the strategy is described which uses the accumulation abilities of the tanks in the ECCS system. In addition, it is considered to use the option to fill SFSP with coolant from the side unit or through LFRU technology. However, the procedures for this are not yet prepared.

The issue of cooling SFSP or the release of the coolant from the TG cooling circuit is resolved within EOPs. The residual output of the fuel can be collected into the ECCS tanks for a certain time. The accumulation abilities when the ECCS tanks are full last for about 4 days.

The deep under-criticality of the spent fuel in the storage pool is ensured by the coolant with the concentration of boron 12 g/kg and by the use of borated steel in the storage grids construction. The use of the borated steel ensures under-criticality in the case that the spent fuel would be cooled by clean water. In addition, in the case of boiling, only water would evaporate (only a slight loss of boron in the escaping steam is expected) and, therefore, any filling with clean water does not lead to a significant decrease in the concentration of boracic acid in the coolant.

Alternatively, it is considered to add coolant from the bubbler channel, coolant from the surrounding unit and the pouring of SFSP from the reactor hall using LFRU. However, the procedures have not been prepared yet.

After interruption of the heat removal from SFSP there would be a permanent increase in the temperature which would be important in the case of filling the upper grid. Without the restoration of the heat removal, first of all there would be the opening of the fuel in the upper layer with the consequent risk of breaking the coverage and the melting of the fuel in the early phase in hermetically separated premises (only the shell of the reactor building), then there would be the release of radioactive products into the surroundings of the NPP.

Due to the fact that the storage pools are not located in hermetically separated premises (only the shell of the reactor building), there would be the release of radioactive products into the surroundings of the NPP. During the origination of a steam-zircon reaction, hydrogen would release into the area of the reactor hall.

Due to the existence of the alternative manner of the heat removal by means of accumulation in ECCS tanks, the long-term loss of the heat removal from SFSP is not expected because in terms of the time for performing the activity for restoration of cooling of the spent fuel stored in SFSP, the situation is more favorable than in the case of the loss of the heat removal from RC. Therefore, no analyses were performed of the damage to the spent fuel stored in SFSP.

In the existing status of the project there are no alternative cooling systems available for filling the coolant into SFSP.

Hydrogen management

The risk of hydrogen in the reactor hall, i.e. outside the containment, was evaluated and it results from the analyses that the failure of the cooling in both pools would probably not lead to such concentration of hydrogen in the reactor hall which would achieve the limits for burning the hydrogen.

The accumulation of hydrogen in other premises is theoretically possible although it was not quantified in detail because a large part of the hydrogen would burn directly in the containment.

Providing adequate shielding against radiation

Due to partial damage to the fuel there would be the probable contamination of the reactor hall of both units by radioactive products released from the melted fuel located in the SFSP. In this case, the reactor hall becomes less accessible and the activities for alternative cooling of SFSP by LFRU would be more difficult. In modes 6 and 7, when the containment is interconnected with the reactor hall, there could be parallel contamination and more difficult access to the containment.

Restricting releases after severe damage of spent fuel in the fuel storage pools

The procedures for the solution to accidents related to melting of the fuel in SFSP are not available yet. The personnel of CR or TSC do not have manuals for Shutdown SAMG (SAMG for shutdown statuses). Nevertheless, the accessible options are known and are based on the continuation of adding water and heat removal and possibly insulation of the release from SFSP according to the EOPs regulation. Damage could take place after a relatively long time with the exception of mode 7 which provides sufficient time for the operative solution.

The principal measure for the restriction of releases into the surroundings or the slowing down of the accident is by filling SFSP with water. The emergency plan for filling pools with water is prepared which will be harmonized with further measures in the reactor hall excluding the presence of operators.

The reactor hall has a large volume which has a positive influence on the dilution of fission products. Further possible measures restricting the release are as follows:

In the case of the release of the activity from SFSP (or from the reactor in mode 6) it is immediately necessary to disconnect the large capacity systems for the ventilation of the hall; this procedure is mentioned in the existing EOP for shutdown statuses.
After all personnel leave the reactor hall, it is important to close all access for personnel into the reactor hall.

In the case that this unit is in mode 6 or 7, i.e. during the replacement of the fuel or during full removing of the fuel from the reactor, where, as a rule, the containment is linked with the reactor hall by further passages, it is necessary to disconnect the ventilation systems for containment, to ensure the evacuation of all persons from the containment and to quickly close all access into the containment of the unit in mode 6 or 7. These measures are enforced by the fact that it is not possible to quickly separate the containment from the reactor hall.

Instrumentation needed to monitor the spent fuel status and to manage the accident

The measurements characterizing the status of SFSP (temperature, level, flow TG) are available only on CR panels. The measurement of parameters related to the cooling of SFSP is not installed to ECR nor is it available in PAMS. Similarly, there is no available PAMS measurement of the Ra situation in the hall near SFSP.

Due to large volume of the reactor hall, its lower tightness and the low residual output of the fuel, no negative conditions inside the containment are expected. Most measurements will remain accessible. The most important are the activities in the atmosphere and the water levels in the SFSP.

Availability and habitability of the control room

The CR and ECR rooms are located in rooms neighboring the containment. This part could be affected by radiation in the case of higher pressure and, at the same time, higher doses inside the containment or in the case of large fission releases from the containment. The actions taken according to EOPs or according to instructions of TSC during the entrance into SAMG assume maintaining the occupation of CR. The transfer of CR personnel to ECR is decided, in special cases, by SE or safety supervisor or RUS. The use of respiration devices at CR is in the competence of RUS.

Before termination of resisting of CR, ECR it is necessary to consider temporary evacuation on the basis of the command of the ERB commander on the basis of the creation of the radiation situation during the fulfillment of criteria in the action instruction (then it could be considered only short-term entrance for carrying out actions).

The direct influence of an accident in the containment on the occupation of CR of the affected unit by penetration through the wall of the containment into CR and ECR will be evaluated.

Access to the reactor hall is possible in the beginning phase of the loss of the cooling of SFSP up to the beginning of the boiling and the decrease of the temperature in SFSP. Due to the decrease of the level in the storage pool and the boiling of the coolant, there would be an increase of the dose rate near the pool and the surroundings would be unacceptable.

The radioactivity would gradually spread throughout the whole hall and into the area of the neighboring unit and all personnel would be forced to evacuate the reactor hall.

Due to the partial damage to the fuel, there would be contamination of the reactor hall and both units by the radioactivity and it would be inaccessible. In modes 6 and 7 the containment can be connected with the reactor hall, therefore, in these modes there would be parallel contamination and inaccessibility of the containment. The occupation of all CR could be indirectly influenced due to the release from the reactor hall into the surroundings and the suction of the ventilation activity or by direct radiation from the area in front of CR.

II.6.4.3 Conclusion on the adequacy of measures to restrict radioactive releases

Even if the prevention of the loss of integrity of the containment as the last barrier against the release of fission products into the surroundings together with the restriction of the release of fission products is the main SAMG objective, SAMG also includes descriptions of strategies for the termination, or at least the decrease, of the release of fission products after the loss of integrity of the containment which use all available means.

The respective manuals will be prepared for more systematic use of all available options for the restriction of the releases from the reactor hall.

II.6.4.4 Measures that can be envisaged to enhance the capability of restricting radioactive releases

Even though there are several diversion systems for the implementation of each strategy for managing accidents, in the areas of the abilities to manage severe accidents, opportunities for further increasing safety were identified. In the area of technical preparedness this concerns the sufficiency of alternative hardware for ensuring the fulfillment of safety functions during the loss of all SSC. The increase of resistance of the NPP Dukovany project to the consequences of severe accidents was decided within PSR. In the area of the technical solution, the increase in the capacity of the existing system for the liquidation of emergency hydrogen is prepared. This relates to the fact that with the existing project capabilities, it is not possible to fully exclude the possibility of the threat to the integrity of the containment by hydrogen in the case of a severe accident. Similarly, the project is developed that prevents damage to the containment by the melt originated during a severe accident; most measures for ensuring the possibility of external flooding of RPV are not yet finished.

In the area of administrative management, the manual for managing severe accidents for shutdown statuses and cases of damage to the fuel in the SFSP is not finished yet.

The opportunities for improvement of the in-depth protection during events where the consequence can be the origination of a severe accident are mentioned in the following table. All of the measures (in the note indicated as “Finding PSR”) would also be implemented without this target oriented evaluation which by its outputs confirms the efficiency and correctness of the previously accepted decisions for the implementation of measures to make the original project more resistant.

Tab. 22: The opportunities for improvement of the in-depth protection for hydrogen management

Opportunity for improvement	Corrective measure	Deadline (short-term I / medium-term II)	Note
Integrity of the containment during a severe accident.	Increase of the capacity of the system for the liquidation of emergency hydrogen.	II	PSR finding

Localization of reactor core melt	Cooling of the melt from the outside of RPV	II	PSR finding
Regulations	Prepare “shutdown SAMG” for shutdown / Severe accident in SFSP	I	PSR finding

III Temelín NPP

III.1 General data about the sites and nuclear power plants

III.1.1 Brief description of the site characteristics

The Temelín Nuclear Power Plant (Temelín NPP) is located in South Bohemia, about 25 km north of České Budějovice, 507,3 meters above sea level.

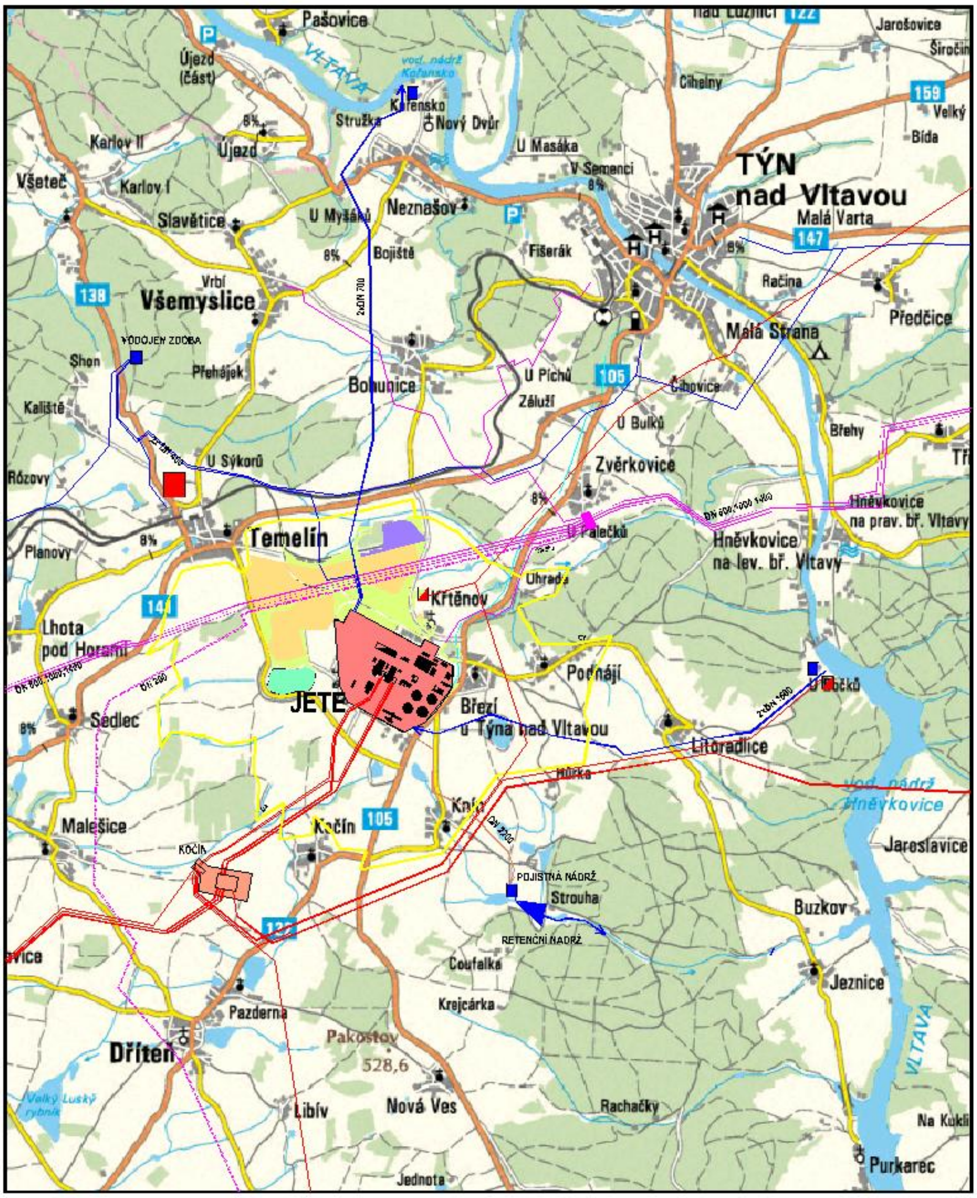


Fig. 23: Location of the Temelín NPP

The nuclear power plant consists of two units containing pressurized water reactors. The nearest town is Týn nad Vltavou, located 5 km northeast of the power plant. The power plant uses water for technological purposes from the Hněvkovice water reservoir on the Vltava river, about 5 km east of the plant. The ultimate heat sink is the atmosphere.

The compound of the NPP includes New Fuel Storage (NFS) and Spent Fuel Storage (SFS). Spent nuclear fuel is stored in a CASTOR-type cask system cooled by natural air circulation within the NFS. Thanks to the passive cooling of the cask system, there is no danger of security function failure in case of an initiating event; therefore, the NFS is not a subject of this safety and safety reserve evaluation.

The holder of operational licence for all nuclear facilities within the site is ČEZ a.s., Duhová 2/1444, 140 53 Prague 4. Currently valid licences were issued for the first unit by a Decision of State Office for Nuclear Safety (SÚJB) on October 4 2010 and for 2nd unit on 11 October 2004.The licences are valid for 10 years.

III.1.1.1 Main characteristics of the units

The nuclear power plant consists of two units with VVER-1000 pressurized water reactors, series type V 320. Each reactor has a nominal output of 3,000 MWt. The primary circuit consists of the reactor, volume compensator and four cooling circulation loops, each containing a main circulator pump and a horizontal-type steam generator.

The primary circuit system is enclosed in a full pressure containment which was built using prestressed concrete. The containment is a cylindrical construction with an internal diameter of 45 m, closed with a semi-spherical cap. The inner surface of the protective containment is hermetically-sealed steel lining. Also located inside the containment are the spent fuel storage pools, where spent fuel is stored after being transported from the reactor core. After the residual heat output is reduced, the spent fuel is placed into the cask system and transported to the Spent Fuel Storage (capacity sufficient for the entire lifetime of the power plant).

The reactor core (active zone of the reactor) is cooled and moderated using light water in the primary circuit, which flows through the reactor core driven by the main circulation pumps. After the coolant flows through the reactor, the heat accumulated in it is transferred via the steam generators to the water in the second circuit. The pressure in the primary circuit is controlled using a pressuriser.

The secondary circuit consists of a steam generator (secondary SG side), water supply system, a turbo generator with a nominal electric output of 1,000 MWe, and a regeneration system.

The active safety systems have 3 x 100% redundancy; they are independent and physically separate. The passive safety systems (hydro accumulators inside the containment) have 2 x 100% redundancy. The seismic resilience of all redundant safety systems, including power supply, control systems and all auxiliary systems, is ensured. The backup power sources and control systems are independent, physically separate and seismically resilient (subject to qualification as safety systems). There are also backup, non-seismically resilient power sources for safety-related systems. The project uses diverse systems to guarantee three basic safety functions: 1) shutdown of the reactor (sub-criticality), 2) transfer of heat (cooling) and 3) preventing leaks (barriers and insulations for the containment)

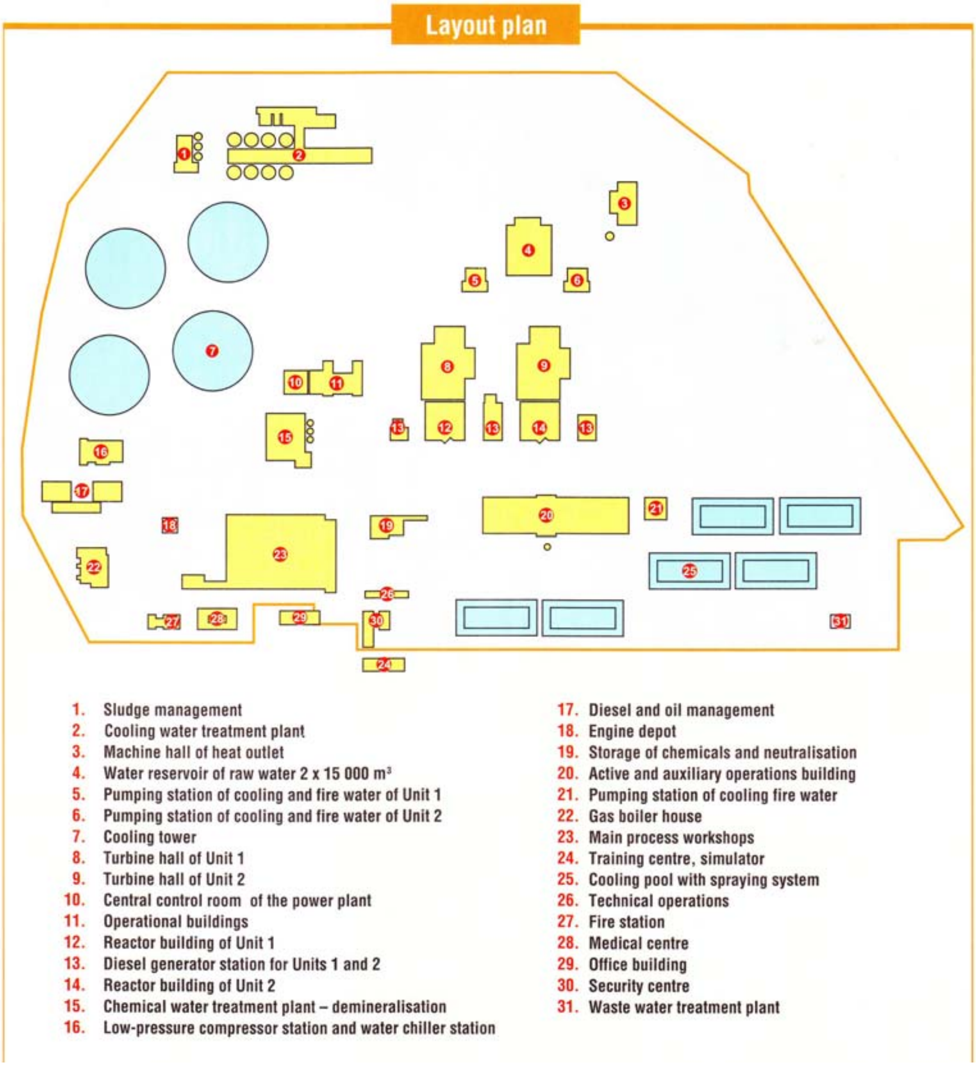
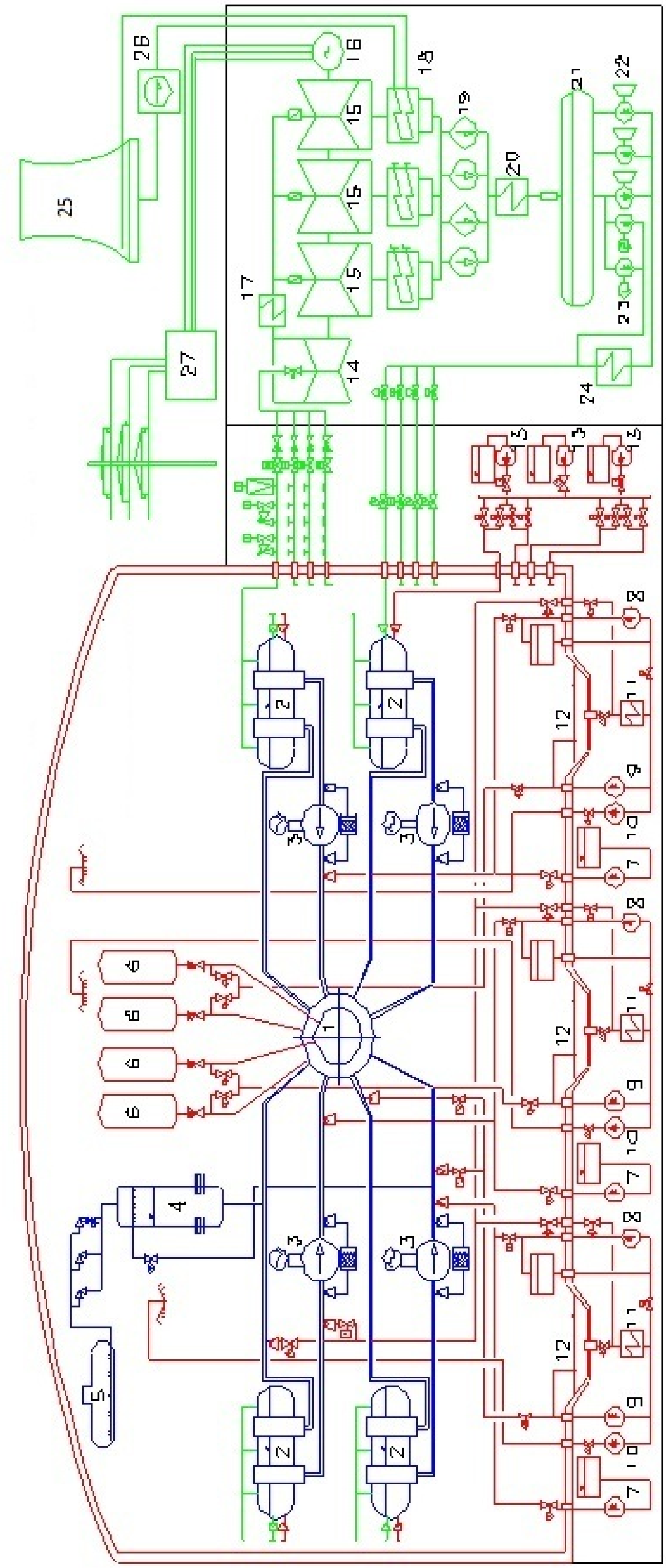


Fig. 24: Main buildings and systems of NPP Temelín



1	Reactor pressure vessel	10	Spray pump	19	Condensate pump
2	Steam generators	11	Heat exchanger	20	LP heater
3	Main coolant pump	12	Containment sump	21	FW tank
4	Pressurizer	13	EFW pump	22	Main FW pumps
5	Relief tank	14	Turbine – HP part	23	Auxiliary FW pump
6	Hydro accumulators	15	Turbnie – LP parts	24	HP heater
7	Hi pressure safety pump	16	Main generator	25	Cooling towers
8	Hi-hi pressure safety pump	17	Separator - reheater	26	Pump station
9	RHR pump	18	Main condenser	27	Main transformer

Fig. 25: Technological scheme of the Temelín NPP

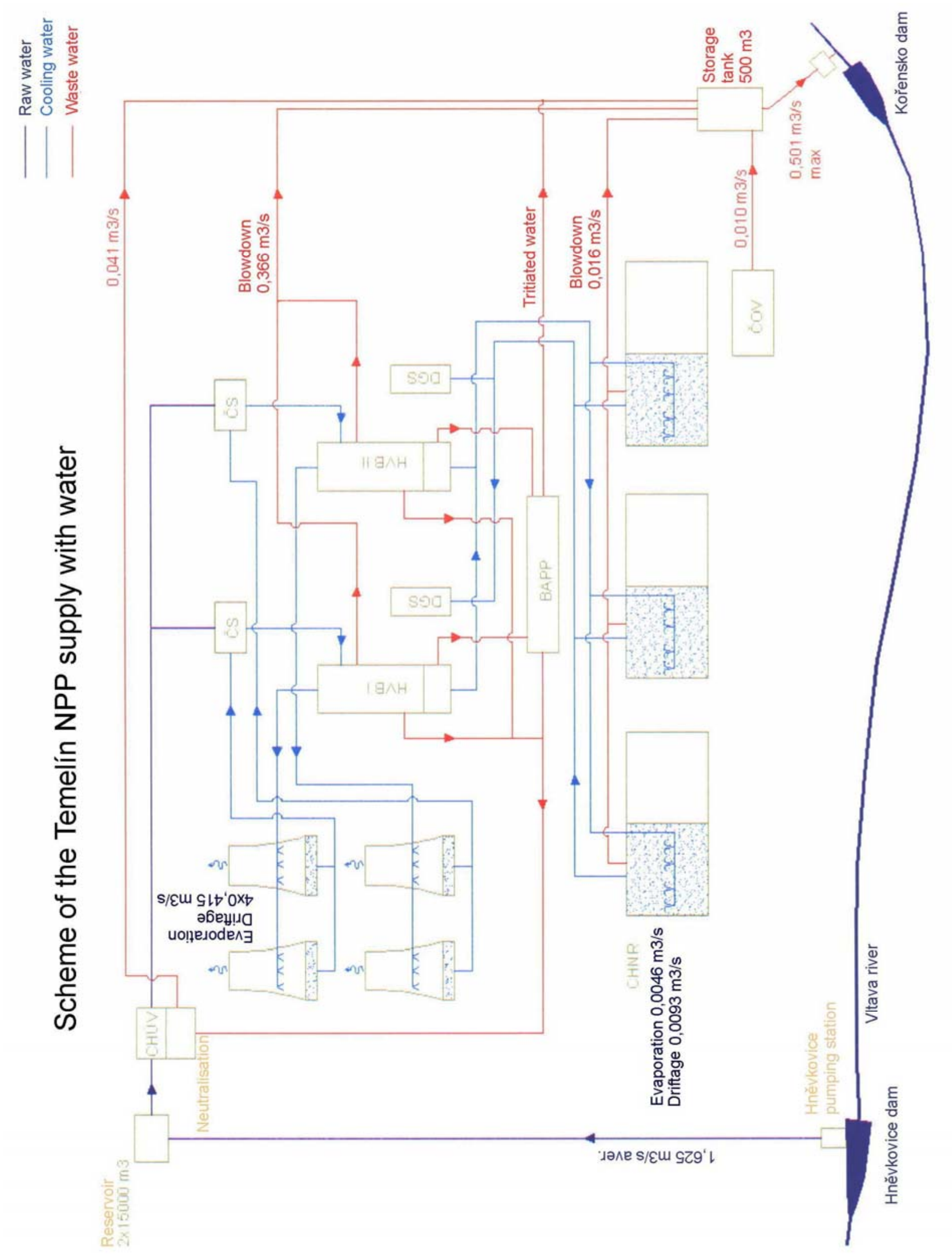


Fig. 26: NPP water supply system

For technological purposes, the power plant uses water from the Hněvkovice reservoir on the Vltava river (approximately 5 km east of the location). The ultimate heat sink is the atmosphere. Under normal operation, residual heat is transferred into the atmosphere via cooling towers (two for each unit); in emergency situations via the steam generators and steam discharge atmosphere stations or via the service water system (ESW) and cooling basins with sprinkler systems (CBSS).

The power plant is connected to the external electrical grid by two 400 kV lines and two 110 kV lines leading through the Kočín switch yard .

Controlled fission (critical state) was first achieved in the first unit on October 11, 2000, and in the second unit, on May 31, 2002.

III.1.1.1.1 Safety key modifications

Based on the results of independent international missions, expert reports and audits, proposals of Czech specialists and requirements of the SÚJB technical modifications to the original projects were proposed to comply with the standards applied to NPP in the west. Key modifications include:

- Change of the control system, including a new project (Westinghouse).
- Change of the original radiation monitoring system, including a project.
- Change and extension of the diagnostic system.
- Replacement of the original cables for non-inflammable and flame-retardant cables.
- Significant modifications in the electrical part (electrical protection, new common diesel generators, increased capacity of accumulator batteries etc.).
- Protection of the high-energy pipeline at +28.8 m – installation of pipe wipe limiters, application of the “no break zone/superpipe” concept.
- Qualification of components crucial for safety, in particular safety valves (functional qualification SBSA, qualification SGSV - steam generator safety valve for water flow and flow of steam-air mix, qualification for pressuriser safety valve and relieve valve for working with water).
- Integrity of the reactor pressure vessel (RPV) , long-term evaluation program using samples from RPV, analysis of pressure-temperature shocks.
- Integrity of the primary loop components – qualification of non-destructive inspections.
- Installation of local seismologic network.
- Implementation of the conceptual solution for accidents beyond design basis.
- Protection against overpressure in the I.C in the cold state.

III.1.1.1.2 Periodic Safety Review

A PSR was carried out for the Temelín NPP in the period between September 2008 and September 2009. Processing procedure PSR ETE (PSR for the NPP Temelín) was implemented to comply with the safety guide IAEA NS-G-2.10 and the WENRA principles for PSR defined in the document titled “Reactor Safety Reference Levels”. PSR NPP Dukovany (PSR for the Dukovany NPP) was carried out by the similar way.

The purpose of the PSR, according to the IAEA NS-G-2.10 safety guide, was to assess the key safety aspects and thus:

- Determine to what extent the power plant complies with current internationally accepted safety standards and practices.
- Verify the validity of the license documentation.

- Determine whether adequate measures for maintaining safety within the power plant are being applied until the next PSR.
- Point out safety areas that should be improved in order to remove the identified safety deficiencies.

Prior to the PSR, the necessary documentation was prepared, which included in particular Methodologies and Criteria based on the legislative requirements within the CR and IAEA documents up to the level of the Safety Guide, INSAG documents and WENRA requirements for the PSR. The evaluation included all 14 areas and was carried out for all safety factors, defined according to the IAEA NS-G-2.10 safety guide.

Comprehensive evaluation within the PSR identified opportunities to increase safety similar to those included in this report. Some of these are in preparation today and they will be implemented regardless of this new assessment; in particular, improvement of the resilience of the Temelín NPP in case of severe accidents (which includes increasing the capacity of the hydrogen disposal system, localization of melt from the reactor core outside the reactor pressure vessel and an alternative water supply to the containment sump). As for the administrative and personnel area, the goal is to create SAMG and PSA Level 2 for shutdown modes and modify the simulator to allow simulating the transition conditions from emergency to the first phases of a severe accident as a tool for training the personnel. The proposed measures in this area are expected to be implemented in 2018, at the latest (until the next PSR).

III.1.1.1.3 Description od the main safety systems

The safety of the Temelín NPP is guaranteed by the ability to fulfill the basic safety functions:

- Safe shut down the reactor and keep it in the state of safe shutdown.
- Transfer residual heat from the reactor core and from the stored spent fuel.
- Keep radioactive leaks within the specified limits.

All SSC are classified according to IAEA standards as safety systems (SS), safety related systems (SRS) or non safety related systems (NSRS).

From the point of view of nuclear safety, the SSC within the Temelín NPP are classified in these categories:

- Important from the point of view of nuclear safety (safety relevant – contribute to safety functions)
- Unimportant from the point of view of nuclear safety (safety irrelevant).

Safety relevant systems are divided into categories based on their functions and relevance for nuclear safety:

- Safety systems
- Safety related systems

“Safety systems” is a set of systems that includes:

- Key protection and key control systems (systems that measure and monitor the parameters and states important for safety and automatically launch safety systems with the aim of securing the unit and keeping it in a safe condition).
- Operating safety systems (systems that carry out safety functions by activating the corresponding safety systems).

- Auxiliary systems (systems ensuring the functions of safety and operating safety systems, such as power supply, cooling etc.).

"Systems related to nuclear safety" is a set of systems that includes:

- Protection and control systems.
- Operating safety systems and constructions.
- Auxiliary systems (power supply, cooling etc.).

Non safety related systems do not fulfill any safety functions; however they can (if available) be used in case of an emergency.

Systems relevant for nuclear safety, i.e. safety systems and safety related systems, are classified systems (classified equipment) and in accordance with legislative requirements, they are divided into three safety classes, depending on their importance for safety.

Technological systems, constructions and components are also classified from the point of view of seismic resilience. All safety systems (and some safety related systems) are included in the first category of seismic resilience.

Safety systems are subject to a qualification process.

Active safety systems are divided into 3 divisions, which are backed up as a whole (concept with 3 x 100% redundancy). In accordance with this concept, each division contains a secure power supply system, which serves as a support safety system for supplying appliances within the corresponding division. In order to achieve the necessary degree of redundancy, the safety systems are independent and physically separate (constructionally, by fire protection), have a separate power source and are controlled separately by the control system. Each system has its own emergency source (diesel generator, batteries), as well as power mains. Thanks to the principle of independence and separation, simple failure within one system will not affect the functioning of the remaining two divisions.

Passive safety systems are based on the assumption that passive system components may fail and therefore this system also has 100 + 100% redundancy. It consists of two independent, functionally identical, but physically separate subsystems.

Units of this type are typical for their ability to provide basic safety functions using the following multiple diversified systems in normal and abnormal modes, as well as in emergencies:

- In power operation mode (ordinary and extraordinary conditions), sub-criticality is ensured by passive and active means – mechanical regulation (clusters falling into the reactor core by their own weight) and by systems for makeup and boron regulation (safety related systems). For emergency conditions and when the above systems cannot be used, it is possible to use safety systems, high-pressure active safety systems with a high concentration of boric acid, low-pressure active and passive safety systems with an shut-down concentration of boric acid.
- While in power operation mode (normal and abnormal conditions) the heat is transferred by horizontal steam generators with large reserves of water, which produce steam for the turbine. During reduced power operation modes, the heat is transferred by stations discharging heat into the condenser, reduction stations, and technological condensers – all included in the safety related systems. From the turbine condensers, heat is transferred into the ultimate heat sink (the atmosphere) by a circuit of cooling water and cooling towers, which create natural circulation. During emergency conditions and when the above systems cannot be used, heat can be transferred by safety systems – stations releasing steam into the atmosphere, as well as by the safety valves of the steam generators - the low-pressure active system

of emergency cooling with heat transfer using ESW with 3 x 100% redundancy with internal 100 + 100% redundancy of the active elements (pumps) inside each of the three equivalent divisions. Heat is transferred from the ESW system into the atmosphere using cooling tanks with sprinklers, also with 3 x 100% redundancy.

- Fission products are prevented from leaking out of the reactor core by physical barriers – matrix and fuel cladding,pressure boundary of the primary circuit, full-pressure containment (c. 60,000 m^3), and by maintaining underpressure. Under ordinary and extraordinary conditions, heat is transferred from the spent fuel pool by a redundant (3 x 100%) system for transferring heat from the SFSP - spent fuel storage pools (safety related systems). As an alternative, in case this system is not available, the heat is transferred by a safety system supplying coolant from the safety spray system with vaporization into the containment. Under emergency conditions (in case of LOCA or HELB), the containment is insulated from the surrounding environment by closing fast acting valves at the borders of the containment; heat is transferred away and pressure in the containment is reduced by active safety spray systems (with 3 x 100% redundancy). Heat is transferred using the ESW into the cooling tanks with sprinklers.

Additional details, including procedures for dealing with beyond design basis accidents, can be found in the following chapters.

III.1.1.1.3.1 Reactivity controls

Sub-criticality of the reactor core

The reactor core was designed to allow the resulting effect of immediate feedback in the reactor core to act against any quick increase in reactivity in all modes of operation with a critical reactor. The reactor and the active zone were designed to work with a negative value of the temperature coefficient of the reactor. Increasing temperature of the moderator leads to decreasing reactivity of the system, which means that the temperature of the moderator has the tendency to return to its original value. An increase in reactivity causing an increase in the temperature of the moderator will thus be limited and will lead to stable conditions of operation.

<u>Operating systems for controlling the reactivity</u>

There are two independent reactivity-controlling systems based on different technical principles:

- Mechanical system for controlling and shutting down the reactor, including a power supply for the drives of the regulating elements.
- System for normal supply and regulation, which uses boric acid.

The mechanical system for controlling and shutting down the reactor is classified as SS.

The reactor protection ensures the activation of release mechanisms, which drop all clusters into the reactor core, aiming to stop the chain reaction.

The mechanisms releasing clusters are triggered automatically after being disconnected from the power supply. A cluster is a bundle of 18 neutron-absorbing rods.

The clusters fulfill the following functions:

- Guarantee quick interruption of the chain reaction in the reactor after being dropped into the reactor core.
- contribute to automatic regulation, which aims at maintaining the reactor output at the desired level and helps during transitions between different output levels.

- compensate for quick changes in reactivity (output and temperature effect).

In total, 61 clusters divided into 10 groups are used. The clusters are divided into 6 groups for shutdown of the reactor (during operation, these are completely withdrawn) and 4 regulation groups (moving by overlapping, and can be controlled automatically or manually). All groups are used for quick shutdown of the reactor when dropped into the reactor core, requiring only the effect of gravity. The system consists of the following main parts:

- Switches (alternating and direct current) for power supply; part of the distributor.
- Clusters.

The clusters are located in a pressurized water reactor in the reactor building inside the containment; the switches are located in the electrical switchboard in the hall surrounding the reactor buiding.

The system for normal make-up and regulation of I.C and boric acid regulation is a system related to safety.

This system is used for makeup the I.C and regulating the concentration of boric acid in the I.C coolant. In order to reduce the concentration of boric acid in the I.C part, the coolant is led through filters and a degasifier into pools with impure condensate. I.C is makeup using clean condensate via a degasifier of boric acid regulation. Increasing the concentration of boric acid in the I.C requires the boron concentrate supply to be connected to the suction intakes of the makeup pumps. Makeup water is heated in the degasifiers using heating steam from the machine room. The system also removes the coolant from the I.C system for cooling and cleaning, before it can be returned to the I.C via degasifier.

The system for draining and makeup is a single technological circuit. The makeup aggregates have 3 x 100% redundancy; two aggregates are connected to a secure power supply (each from a different common DG).

The system and pipes are inside and outside of the containment. A regeneration exchanger of I.C water and an additional cooler are located inside the containment. Makeup aggregates, a makeup degasifier, a boron regulation degasifier and a degasifier steam cooler are located outside of the containment. Just like the makeup water cooler, additional coolers and the condensate cooler are located outside the containment.

Makeup aggregates, consisting of two serially connected pumps, are located in separate rooms; therefore, they are physically separate.

In order to function, the system for normal makeup needs an auxiliary system – an oil management system. Each makeup and boron regulation pump has an independent oil-management system, which includes pumps, coolers, filters and a tank. Oil is led from the oil tank into coolers by pumps. From the cooler, some oil goes directly to the hydraulic clutch of the pump and some via a filter to the bearings and gear, then back to the tank. The oil management system for makeup pumps is located in the reactor building outside the containment. Each aggregate has an oil management system in a separate room, which means that the oil systems are independent. The power supply for the oil pumps is similar to that of the makeup aggregates.

Additional auxiliary and support systems provide makeup of boric acid for the I.C; in particular, there is a system for supplying boric acid, which is also classified as a safety related system.

The purpose of this system is to store H_3BO_3 concentrate and supply it into the primary circuit. The system consists of two storage tanks with H_3BO_3 concentrate, and three pumps. H_3BO_3 concentrate from the concentrate cleaning station, as well as from the chemical reagent preparation, is led by independent routes into each of the two supplying tanks. The concentrate (40 g H_3BO_3/kg) is supplied from the tank into the suction openings of the makeup pumps.

Alternatively, it is also possible to bring a solution of boric acid from impure condensate tanks to the suction opening of the makeup pumps via a degasifier of the standard makeup system.

Tanks and pumps for boron concentrate are located in rooms outside the containment.

Safety systems

In addition to the operating systems that control reactivity, the reactivity is controlled by a high-pressure system for emergency cooling of the reactor core and by the system for emergency boroning; both are classified as safety systems.

These systems are used to limit the development and mitigate the consequences of accidents that lead to a loss of tightness in I.C, and possibly also in II.C (LOCA and HELB). Under normal operation of the unit with nominal or reduced output, the system is in standby mode, ready to intervene in case of an emergency. In an emergency, this system prevents undesired transitional processes connected with changes in reactivity.

All supporting systems have 3 x 100% redundancy (cooling, power supply, control and ventilation).

A high-pressure system for the emergency injection of boron consists of three piston pumps and the high-pressure system of emergency cooling of the reactor core consists of three centrifugal pumps. Each pump has own tank with a supply of a concentrated boric acid solution (40 g/kg). All pumps are connected to a secure power supply (DG).

The system is located in the reactor hall outside the containment with pipes leading into the containment, where they are connected to the primary circuit. The pumps are located in separate rooms.

The tanks of the high-pressure emergency makeup system are located inside the containment in separated premises. The high-pressure emergency boron injection tanks are located outside the containment in separate rooms.

Besides these methods of ensuring sub-criticality of the reactor core, there is an additional reserve of boric acid solution in the containment sump-tank, common for the high-pressure, low-pressure ECCS and containment spray systems, available in case of emergency. A sufficient concentration of boric acid is also available in hydro-accumulators, constituting a passive system of emergency cooling for the reactor core.

Sub-criticality of the spent fuel pool storage

The sub-criticality of spent fuel sets stored in the spent fuel pool is guaranteed by two independent methods:

- Geometry and material used for storage bars located in the spent fuel pool.
- Concentration of boric acid in the pool.

Sub-criticality is also guaranteed in case the spent fuel pool is filled with pure condensate.

III.1.1.1.3.2 Heat transfer from the reactor to the ultimate heat sink

Tab. 23: The VVER1000 nuclear units in the Temelín NPP have 6 modes of operation

Mode	Name	Thermal output	Average temperature in I.C	k_{ef}
1	At power operation	≥2 % N_{NOM}	> 260 °C	<0.99
2	low power operation	< 2 % N_{NOM}	> 260 °C	<0.99
3	Hot state	Residual	> 260 °C	< 0.99
4	Semi-hot state	Residual	260 °C >T_{mid} □□150 °C	< 0.99
5	Cold state	Residual	150 °C > □T_{mid} □□70 °C	< 0.99
6	Shutdown	Residual	< 70 °C	< 0.98

Under a certain configuration of the system with reactor shutdown, the probability of damage to the reactor core can be higher than in full operation. The most hazardous mode of operation of the unit is the following configuration: Loss of systems transferring residual heat in mode 6 with the surface in the reactor along the axis of cold nozzles.

If the primary circuit is under operation with surface in the reactor at the axis of cold nozzles with nuclear fuel in the reactor core, it is necessary to keep cooling the reactor core with an increased intensity. Lowering the surface in the reactor is permitted only after the residual heat output is reduced to a level that can be managed easily using the system for the transfer of residual heat. In this mode of operation (especially shortly after a shutdown of the reactor), the I.C generates considerable residual heat output, while the volume of the circulating coolant is low. There are also certain organizational limitations, such as repairs and revisions within certain systems or the absence of automatic actions of the control system. In such situations, there is only a minimal time reserve for reacting to a cooling failure or uncontrolled loss of coolant and for recovering cooling functions and locating and stopping possible leaks. For these reasons there is a number of technical and administrative limitations in case the surface in the reactor is lower, which aim to minimize the risk that the residual heat transfer fails (making sure there is a sufficient amount of coolant and technical means for makeup I.C, pre-job-briefing before draining the I.C , permanent presence of selected employees in the CR etc.).

The following chapters describe the methods of transferring heat from the reactor core into the ultimate heat sink in Modes 3 through 6, i.e. from the mode in which the unit is shut down, and it is necessary to transfer heat from I.C into the ultimate heat sink to a mode in which it is possible that the I.C will lose its sealing ability and the residual heat must be transferred with a low surface in the reactor.

III.1.1.1.3.3 Existing heat transfer means

Operating systems (classified as non safety related systems or safety related systems) can be used for transferring heat from the reactor core, as well as other systems that are classified in the category of safety systems.

Operating systems transferring heat from the reactor core

The standard operating system designed to transfer heat from the reactor core into the ultimate heat sink is the secondary circuit, which transfers heat from the reactor core is transferred by means of forced circulation (provided that MCP is in operation) or by natural circulation. In the cooling down mode, the steam from the SG is led into the turbine condenser and heat from the condenser is transferred using the cooling water into the cooling towers by natural circulation. The condensate is then returned from the condenser by the main or auxiliary condenser pumps to the tank from which the water is supplied to the SG by the auxiliary or turbo supply pumps. The auxiliary pumps are used for makeup condensate or supplying water in Modes 2 and 3.

During cooling down, after the parameters in the I.C are brought down to values that make the transfer of heat into the turbine condenser inefficient, the heat is transferred using a low-pressure cooling system of the reactor core (safety system), which in this mode works as a standard operating system. Heat is transferred from the I.C via ECCS exchangers, which forward the heat to the ESW and then into the atmosphere via cooling towers with sprinklers. The cold coolant of the I.C is returned from the ECCS exchangers to the I.C in this mode by the ECCS low-pressure pumps.

Safety systems

In case the standard operating systems described above are not available, systems classified in the safety systems category will be used.

Heat is transferred from the I.C using forced or natural circulation of the coolant. In this case, heat from the II.C is transferred using a non-closed circuit, i.e. stations or SFSG (safety valve of the steam generator) discharging heat into the atmosphere. This mode is referred to as secondary “feed and bleed”. If the auxiliary pumps supplying water into the SG fail, water is supplied by emergency pumps (with 3 x 100% redundancy).

If heat cannot be transferred from the I.C via the secondary heat transfer system – release of steam into the atmosphere (via atmosphere heat discharge stations or via the SFSG) or if coolant leaks from the I.C, there is an alternative method of heat transfer from the reactor core; the primary “feed and bleed” method, when coolant is added into the I.C in a controlled manner using the ECCS and drained from the I.C into the containment. In this mode, heat would be transferred via the ECCS exchangers cooled by ESW and via the CBSS into the atmosphere.

Other systems

Using other systems (beyond the scope of their design specification) is described in Chapter III.6, Severe Accident Management.

III.1.1.1.3.4 Layout information on the heat transfer chains

Operating systems

The auxiliary supply pumps system is designed to supply water to the steam generator during activation and shutdown of the unit. It consists of two electrical pumps, which supply water from the supply tank into the SG common supply collector behind the high-pressure supply water heaters. The pumps and valves are connected to a secure power supply (each pump from a different DG). The source of water for the auxiliary supply pumps is the supply tank located between the turbine hall on level +30.0 m.

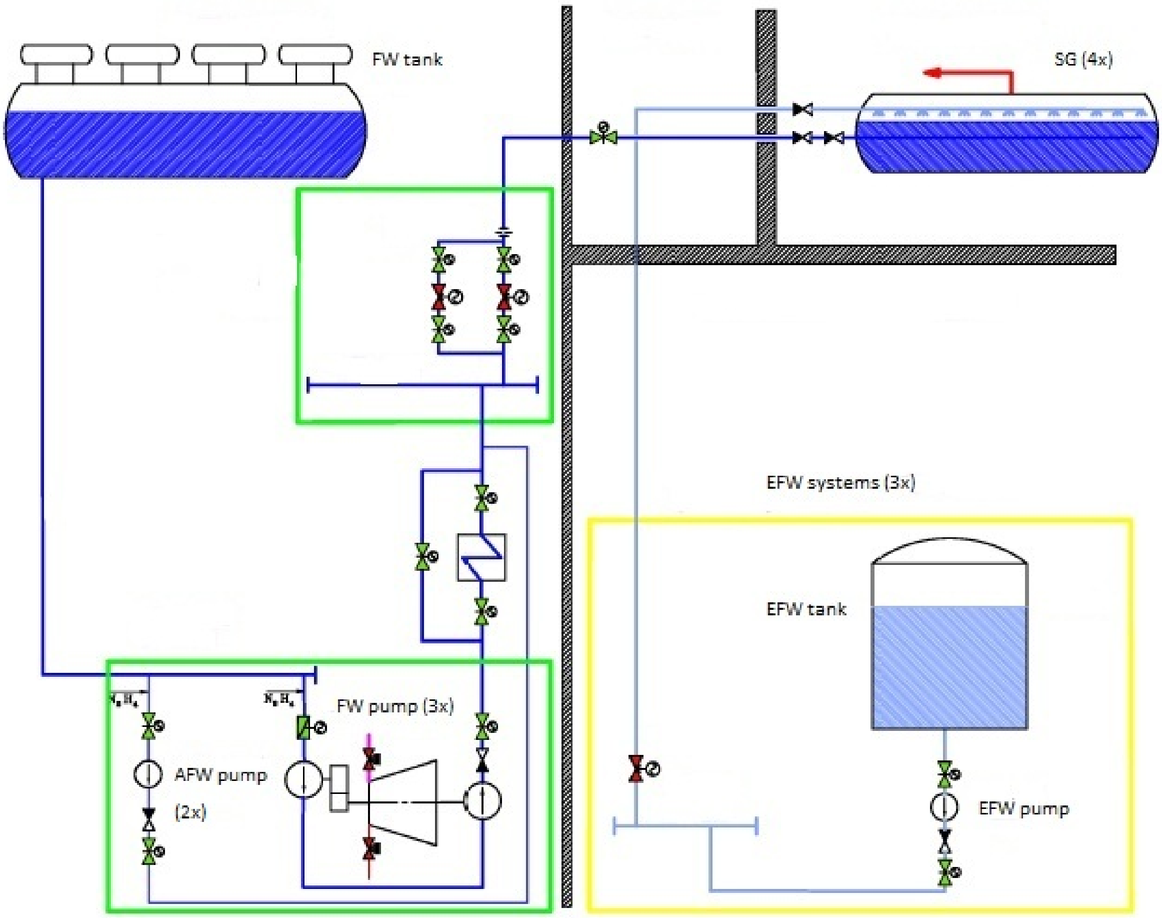


Fig. 27: Overview of steam generator feedwater systems

The system of auxiliary condenser for transferring residual heat is used in case of planned cooling down with a temperature of the I.C lower than 150°C and pressure in the I.C below 1.7 MPa. If the pressure in the I.C is higher than 1.7 MPa, it is impossible to connect the system for standard cooling down, because the permissible parameters for an ECCS exchanger are exceeded. The rate of the cooling down is usually 30 °C/hr; maximum in case of emergency is 60 °C/hr. The residual heat transfer system can be operated in the direct coolant circulation mode, when cold coolant is supplied into the cold branch of the circulation loop/under the reactor core and hot coolant is drained from the hot branch of Loop No. 4 into the ECCS exchanger and on the suction of the low-pressure pump for emergency makeup. In case the coolant level in the reactor is lower, the residual heat transfer system runs in a reverse circulation mode; cold coolant is added to the hot branch of the circulation loop/above the reactor core and hot coolant is drained from the cold branch of Loop No. 4 into the ECCS exchanger (from which the heat is transferred into the ultimate heat sink via the ESW system) and to the suction of the high-pressure pump for emergency makeup.

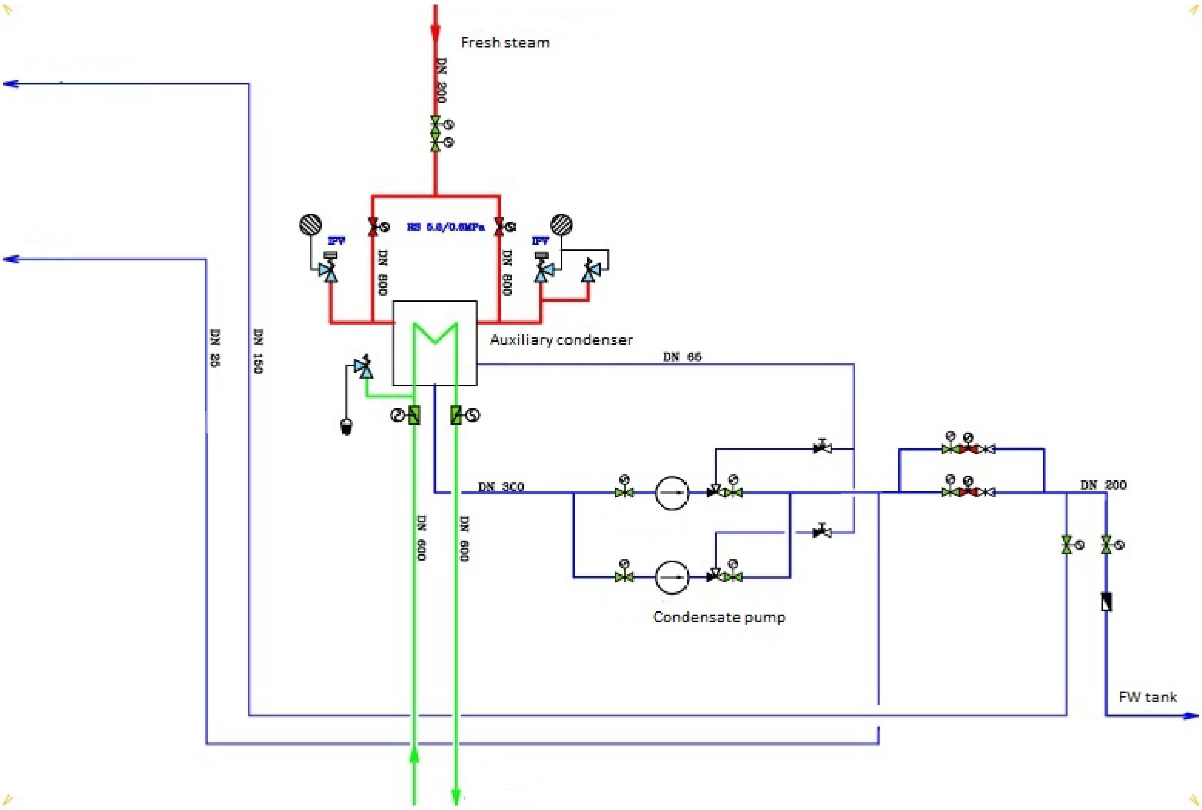


Fig. 28: Auxiliary condenser system

<u>Safety systems</u>

The emergency water supply system for the SG is designed to provide safe water supply for the SG in case the level in two SG decreases below an acceptable level. The system consists of three independent channels (with 3 x 100% redundancy), each of which includes an emergency pump, 500 m^3 tank with demi water and pipes. To enable the water from the tanks to be used by each sub-system, the three tanks are connected by pipes and two pairs of separating valves.

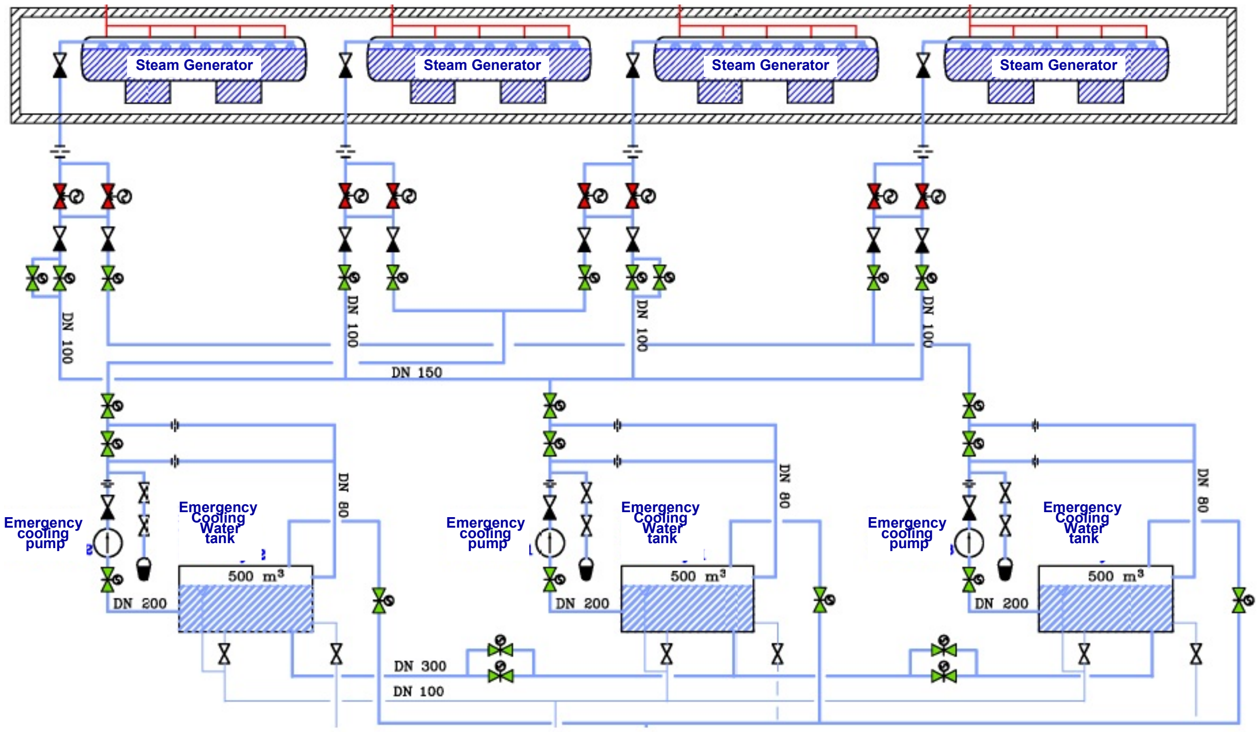


Fig. 29: System of emergency water supply for the SG

The demi water tanks, including their interconnecting pipes, are located in a separate room. The emergency pumps and valves of the emergency water supply are located in the hall surrounding the reactor room, in separate rooms.

The steam bypass stations into the atmosphere carry out the safety function of transferring heat from the SG by releasing steam into the atmosphere during accidents involving a loss of tightness of the SG if the station releasing heat into the TG condenser is insufficient for reducing the pressure and preventing pulsating operation of the security valves in all modes of the unit's operation. The stations releasing heat into the atmosphere are connected to various sections of secure power supply from batteries. The station releasing heat into the atmosphere opens when the specified limits of pressure are exceeded and pressure is maintained by regulating the pressure of steam in the SG. After the pressure is reduced, the releasing station is closed.

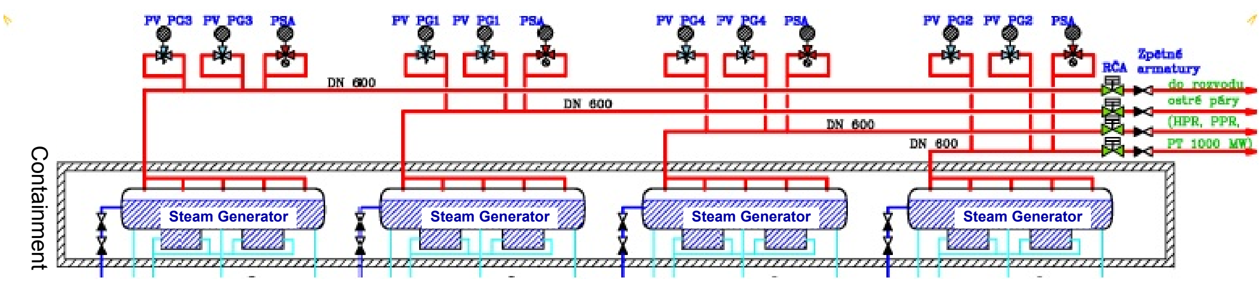


Fig. 30: Main steamlines, SGSV, SBSA

Another layer of protection for the secondary circuit preventing overpressure consists of two impulse **safety valves of the SG**. The SGSV are gradually opened when the pressure exceeds the value specified as the opening pressure for the atmosphere heat discharge station.

The steam bypass station to atmosphere and the SGSV are located in the neghbouring building surrounding the reactor hall, outside the containment.

A high-pressure system of emergency cooling is used for reducing the severity of accidents and removing the consequences of accidents, which include a loss of tightness of the I.C or II.C. During normal operation of the unit at nominal or reduced output the system is in standby mode, ready to be activated in case of an emergency. In terms of heat transfer from the reactor core, the system:

- Supplies a sufficient amount of boric acid (H_3BO_3) into the I.C in case the I.C loses tightness or the II.C breaks. The aim is to prevent fuel degradation.
- In combination with other safety systems, ensures that radioactive leaks and the release of ionizing radiation from the containment in an emergency and after an emergency are minimized.

The high-pressure system has systematic 3 x 100% redundancy, including all support systems (cooling, power supply, control system and ventilation). The high-pressure system of emergency cooling of the reactor core consists of three centrifugal pumps. Each pump has its own tank with a concentrated solution of boric acid (40 g/kg). All pumps are connected to a secure power supply (DG).

The system is located in the reactor chamber inside the containment, and the pressure pipes are led into the containment and connected to the primary circuit. The pumps are located in separate rooms. High-pressure tanks of the emergency makeup are located inside the containment, in separate premises.

Also available for the purposes of long-term cooling of the reactor core in emergency situations is an additional reserve of boric acid in the sump-tank in the containment (c. 12 g/kg), common for the high-pressure, low-pressure ECCS and the containment spray system.

A low-pressure system of emergency cooling is used (besides planned cooling down) for reducing the severity of accidents, dealing with the consequences of accidents involving a significant loss of tightness of the I.C (large LOCA). Under normal operation of the unit with nominal or reduced output, the system is in standby mode, ready to be activated in case of an emergency. In terms of heat transfer from the reactor core, the system:

- Makeup and increases the concentration of boric acid in the I.C in case of a significant loss of tightness and transfers the residual heat from the I.C in an effort to minimize fuel degradation.
- Maintains the reactor in a state of safe shutdown after each shutdown.
- In combination with other safety systems, ensures that radioactive leaks and the release of ionizing radiation from the containment during and after an emergency are minimized.

The low-pressure system has systematic 3 x 100% redundancy, including all support systems (cooling, power supply, control system and ventilation). The high-pressure system for emergency cooling of the reactor core consists of three centrifugal pumps. Boric acid is supplied from a sump-tank in the containment, common for all three divisions of the emergency safety systems (high-pressure, low-pressure ECCS) and spray system in the containment). All pumps are connected to a secure power supply (DG). In emergency modes, as well as in the mode of planned cooling down, coolant for makeup of the I.C is cooled in the ECCS heat exchanger using ESW.

The system is located in the reactor building outside the containment, and the pressure pipes are led into the containment and connected to the primary circuit. The pumps of the low-pressure system are located in separate rooms and the tank (reservoir) is also in an independent room within the containment, with three inlets from the lowest floor of the containment. The heat exchangers of the ECCS are located in three independent and separate rooms outside the containment.

A passive system of emergency cooling of the reactor core (system of hydro-accumulators) is used for quick flooding of the reactor core in emergencies, when the pressure in the I.C drops and a large amount of coolant releases from the reactor. The system includes four hydro-accumulators, which supply a solution of H_3BO_3 (12 g/kg) under and above the reactor core. It is a passive system, which does not need a power supply to function. The system consists of a pressure tank for emergency cooling of the reactor core connected by piping. The stored compressed nitrogen expands and pushes the coolant into the reactor. This system is passive, i.e. if the pressure in the I.C drops below the pressure in the tank, the contents are released under pressure into the I.C. It is activated with no external initialization impulse and it works without a power supply. To prevent nitrogen from entering the I.C, the routes contain valves connected to a secure power supply (battery), which closes the routes after the pressure tanks are emptied.

The nuclear safety concept is based on the assumption that some components of the passive system may fail, and therefore the entire system has 100 + 100% redundancy; it consists of two independent and functionally identical subsystems, each with two pressure tanks.

The passive cooling system of the reactor core is located inside the containment. Pressure tanks and hydro-accumulators are located in pairs in separate premises within the containment.

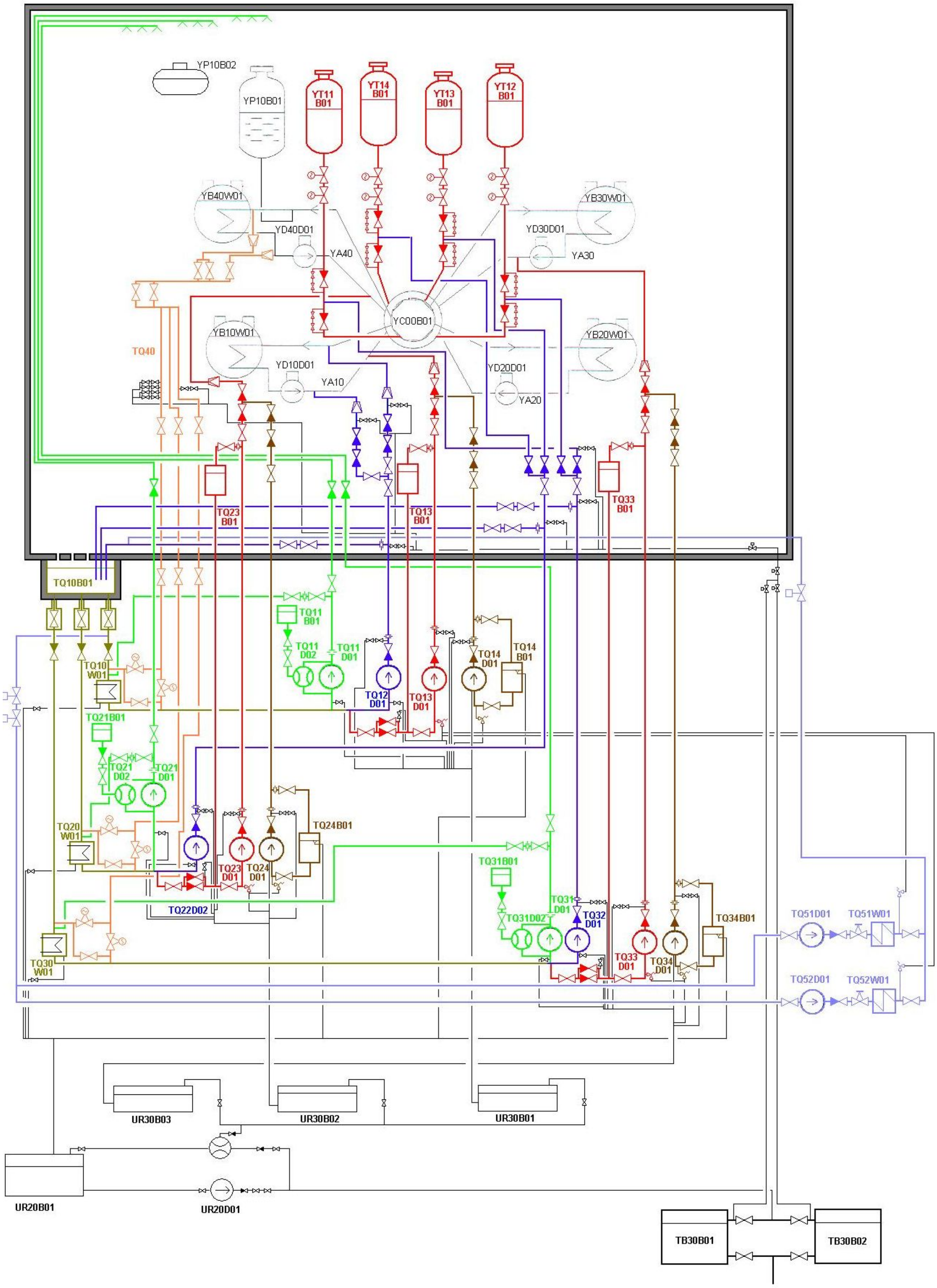


Fig. 31: Overview of emergency systems

Other systems

In the highly improbable situation that the heat transfer capacity from the reactor core is lost (loss of secondary heat transfer systems combined with failure of the primary “feed and bleed” system) there are other strategies available to ensure secondary heat transfer using systems beyond the scope of their projected purpose.

Makeup of the SG using condenser pumps from the supply tank (route designed for rinsing the II.C). If the flow rate of condensate into the SG is to be adjusted, it is necessary first to reduce the pressure in at least one intact (undamaged) SG by releasing steam into the atmosphere to bring the pressure close to the atmospheric pressure with respect to the pressure in condenser pumps. This strategy is described in the EOP.

Another possibility of using a system beyond the scope of its projected purpose is **gravitational filling of the SG from the supply tank**. Because the supply tank is at +30.0 m and the SG at +28.8 m, gravitational filling of the SG from the supply tank first requires the release of steam into the atmosphere and thus reduction of the pressure in at least one intact (undamaged) SG to a value close to the value of the atmospheric pressure. The operating pressure in the supply tank, which is even with the reactor shut down at about 0.6 MPa, is a good starting point allowing gravitational filling of the SG from the supply tank to provide at least partial heat transfer from the I.C. This strategy is described in the SAMG.

III.1.1.1.3.5 Time constraints for the availability of different heat transfer chains

Based on the engineer’s assessment, trivial calculations and comparisons with existing analyses, it is possible to estimate the minimum periods during which the individual systems are capable of transferring heat from the reactor core.

Operating systems

An auxiliary supply system for the SG supplies water into the SG from the supply tank. There is about 350 m^3 of water in the supply tank. The supply tank can also be refilled from the TG condenser, which contains about 250 m^3, or from the demi water tank containing 2 x 770 m^3, common for both units. This amount of water available to be supplied to the SG is sufficient for cooling the unit to a temperature that permits activating the residual heat transfer system and thus bringing the unit to a cold state.

After the parameters of the I.C have dropped (temperature in the I.C lower than 150 °C, pressure in the I.C below 1.7 MPa), the residual heat transfer system is activated to continue cooling the unit until it reaches a cold state throughout the closed circuit (reactor core – exchanger ECCS – low pressure pump for emergency makeup – reactor core). In this cooling mode, it is necessary to add into the I.C only an amount of coolant that compensates for volume changes of the coolant due to temperature changes. For this purpose, there is a sufficient reserve of coolant with a corresponding concentration of boric acid in the reserve tanks of the operating as well as safety systems, and there are means of supplying this coolant to the I.C of the unit using a closed circuit for residual heat transfer, so that the reactor can be kept in a cold state indefinitely.

There are no time constraints for the availability of the residual heat transferring operating systems, and therefore, no time constraints for maintaining the unit in a cold state.

Safety systems

When using emergency supply system for the SG, water is supplied into the SG from the reserve tanks of this system and heat is transferred by releasing steam from the SG into the atmosphere (secondary “feed and bleed”). To replenish demi water in the SG, there are 3 x 500 m^3 reserve tanks of the emergency supply system for the SG available for each unit, as well as 2 x 770 m^3 tanks common for both units. This amount of water is sufficient for cooling the units to a cold state (according to the design, one system of emergency supply for an SG is sufficient for cooling the unit to a cold state) or maintaining unit in a hot state for about 72 hours.

If, for any reason, it is not possible to use the secondary circuit for transferring heat from the reactor core, the systems of reactor core emergency cooling in primary “feed and bleed” mode can still be used either in a hot or semi-hot state. In this mode, coolant is removed from the I.C in a controlled manner (using the system of emergency bleeding of the I.C or using the release valve of the containment) via the condensing tank and into the containment; then via the ECCS exchanger (from which the captured heat is transferred into the ultimate heat sink by the ESW system), it is returned to the I.C using the pump of high-pressure emergency cooling system of the reactor core. The heat released into the containment is, in this case, transferred by the containment spray system to the ECCS exchanger and finally, to the ESW system.

There are no time constraints for heat transfer using the primary “feed and bleed”.

Other systems

A reserve of demi water in 2 x 770 m^3 tanks, common for both units, is available for makeup of the SG using condenser pumps. Provided that the supply of water into the SG using condenser pumps is efficient, this amount of water is sufficient for cooling the unit to a cold state and maintaining it in this state for at least 24 hours, which is sufficient for carrying out other activities to ensure that the heat is transferred from the reactor core by other means.

For gravitational filling of the SG from the supply tank there is 350 m^3 of water available, usually kept in the supply tank. Because this is an emergency solution only to be used when all other means of transferring heat from the reactor core are unavailable, this volume of water only provides several hours of cooling, which should be sufficient for carrying out activities to ensure that the heat is transferred from the reactor core by other means.

III.1.1.1.3.6 AC power sources and batteries

The power supply sources for all systems providing heat transfer from the reactor core have the following defence-in-depth structure:

1) Operating power sources (operating transformers for house consumption), or
2) Backup power supply from the grid, or
3) Secure power supply from systematic and common DG and from batteries.

Operating systems ensuring sub-criticality of the primary circuit, as well as the systems providing the transfer of heat within the secondary circuit, are connected to the system of secure power supply for safety related systems (common DG and accumulator batteries). All safety systems are connected to the system of secure power supply for safety systems (DG and accumulator batteries). If the external power supply is lost and the corresponding DG is activated, the pumps of the systems are activated automatically.

The pumps and their valves, whose positions must be changed in order to fulfill a safety function, are connected to the corresponding DG. Selected key components (valves of atmosphere heat discharge stations, separating valves of hydro-accumulators, fast acting valves for insulating the containment, etc.) and the corresponding OCS systems are connected to batteries.

III.1.1.1.3.7 Need and method of cooling equipment

Operating systems

The unit is cooled using operating systems in two stages. In the first stage, the residual heat from the reactor core and the accumulated heat are transferred via the II.C in the form of steam into the atmosphere, either directly by transferring steam from the SG into the atmosphere, or by transferring steam into the TG condenser and via cooling water into the cooling towers. In the second stage, residual heat from the reactor core and the accumulated heat are transferred via ECCS exchangers by means of the ESW and CBSS into the ultimate heat sink – the atmosphere.

Steam and water modes can be considered to be two physically different methods of cooling down, because they use different systems in operation. However, due to its physical principle, the steam mode cannot cool the unit to the cold state.

Safety systems

Safety systems of the secondary heat transfer (makeup of supply water in the SG using the emergency water supply and steam transfer from the SG) transfer heat in the steam mode directly into the atmosphere. In order to transfer heat from the components of the safety systems (pumps, OCS, unit's control room/emergency control room, etc.), and in case the external power sources and DG power sources are lost, it is necessary to use the ESW system. However, the transfer of heat from the secondary side is simplest and requires minimal resources.

In case the secondary heat transfer system cannot be used for the second stage of cooling by the residual heat transfer system, the residual heat from the reactor core and the accumulated heat are transferred via the ECCS exchangers by means of the ESW and CBSS into the ultimate heat sink – the atmosphere.

Also, for the primary "bleed and feed" it is necessary to run the ESW system, because heat is transferred via the ECCS exchangers, ESW and CBSS into the ultimate heat sink – the atmosphere.

Other systems

When supplying water to the SG using condenser pumps or in case of gravitational filling of the SG, water supplied to the SG is turned into steam, which is then released into the atmosphere via atmosphere heat discharge stations. In fact, this is a passive method of heat transfer because as long as the SG receives a sufficient amount of water for heat transfer from the I.C, the steam is released into the atmosphere by remotely opening the valve on the release station (power supply from batteries) or by manually opening the valve on the spot (the valve is located outside the containment in an accessible part of the hall surrounding the containment).

III.1.1.1.4 Heat transfer from spent fuel pools to the ultimate heat sink

Spent fuel storage pools (SFSP) are located inside the containment and have two sections. From the point of view of residual heat transfer from the SFSP, there are two possible initial states:

- The pool contains spent fuel from previous campaigns in order to reduce its activity and residual heat output.
- The entire reactor core is moved into the SFSP so that, together with spent fuel from the previous campaigns, the SFSP also contains partially spent fuel with a high residual heat output.

As long as the SFSP contains stored fuel, the water surface in the pool is kept higher than 792 cm, which is sufficient for the shielding and cooling functions. The volume of water in the SFSP in this mode is about 223 m^3 in sections 01 and 03 and about 104 m^3 in section 02 (in the refueling mode, these values are doubled). In both cases, heat is transferred by the circulation of water in the SFSP, forced by a dedicated system of SFSP cooling (3 x 100 %) via exchangers cooled using the ESW system, which then transfers heat using the CBSS into the ultimate heat sink – the atmosphere.

III.1.1.1.4.1 Existing heat transfer means

To transfer heat from the SFSP, it is possible to use operating systems (classified as safety related systems) in combination with safety systems, but it is also possible to rely solely on the systems from the category of safety systems.

Operating systems transferring heat from the SFSP

Under normal conditions, the heat from the SFSP is transferred into the ultimate heat sink by an SFSP cooling system, which transfers heat from the SFSP to heat exchangers, from which the heat is transferred using the ESW system (3 x 100%). The ESW system dissipates heat into the atmosphere in the cooling basins with sprinkler systems. The cold coolant is then returned from the exchangers to the SFSP using the SFSP cooling system pumps.

Safety systems

If the operating systems cannot transfer residual heat from fuel in the SFSP, it is possible to transfer heat by adding water to the SFSP using any of the three spray pumps in the containment. The source of coolant is the tank-reservoir in the containment, and the coolant is drained into the containment from the SFSP by gravitational flow, returned to the reservoir in the containment and via the ECCS cooler (from which the extracted heat is transferred into the ultimate heat sink by the ESW system), then returned to the SFSP by a spray pump.

Other systems

If there is no combination of pump-exchanger-ESW subsystems capable of transferring the residual heat from the fuel, it is possible to transfer the heat by increasing vaporization in the containment, with the tank-reservoir in the containment being the source of the coolant that is supplied to the SFSP by any of the three spray systems in the containment.

Another possible way of resupplying the SFSP is to use a pump from the coolant purification system of the SFSP.

III.1.1.1.4.2 Layout information on the heat transfer means

Operating systems for the transfer of heat from the SFSP

Under normal operating conditions, the heat is transferred by one of the three circuits of the SFSP cooling system, which is classified as a safety related system with 3 x 100% redundancy. The coolant is a solution of boric acid, although the SFSP can theoretically be filled also with pure condensate. The cooling circuit is seismically resilient. The capacity of the cooling circuit is sufficient for both initial states of the level in the SFSP.

The coolant is circulated by three independent cooling circuits for each SFSP section. Each circuit has own pump and heat exchanger cooled by the ESW. To increase the reliability and operability of the entire system of heat transfer from the SFSP, these three circuits are linked on the side of pump suction openings, but also on the side of pump pressure outlets, which allows operatively combining the chain of heat transfer (section of the SFSP filled with spent fuel, pump, exchanger with the corresponding ESW system). The pumps are connected to a secure power source (DG). The SFSP cooling system is located in the hall surrounding the reactor building outside the containment; the SFSP is located inside the containment.

Because this is a closed cooling circuit for the SFSP coolant and the SFSP are covered, there is only very limited vaporization; therefore, the SFSP cooling systems are capable of transferring heat from the spent fuel located in the SFSP for a practically unlimited time.

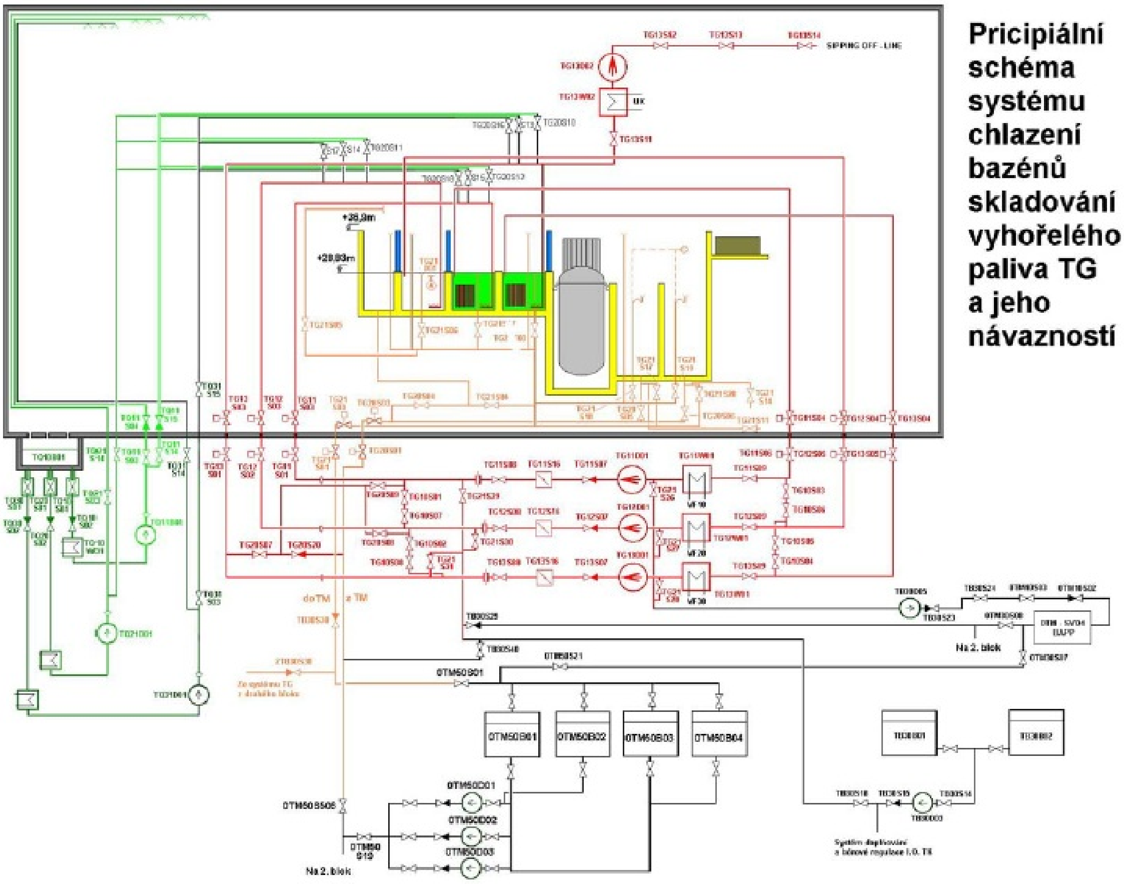


Fig. 32: SFSP system

Safety systems

In case standard cooling of the SFSP is completely lost (whether this is due to a low level or to interruption of heat transfer), the SFSP is supplied using a TQ containment spray system set for emergency resupply of the SFSP. This system allows resupplying of the SFSP and, using gravity, removing the coolant from the SFSP to the reservoir in the containment and thus also transferring heat from the spent fuel in the SFSP in an alternative manner via the ECCS cooler. This cooling circuit is independent of the normal SFSP cooling system and provides an alternative way of transferring heat from the spent fuel stored in the SFSP. However, even in case of emergency cooling, the heat extracted from the SFSP goes through the ECCS exchangers into the ESW.

If it is possible to close the cooling circuit via the reservoir in the containment and ECCS exchanger, this method of cooling the SFSP also guarantees the transfer of heat from spent fuel located in the SFSP for a practically unlimited period of time.

Other systems

In case none of the combinations of forced cooling circuits (pump of the TG, exchanger, ESW) is able to provide the transfer of heat from the SFSP, the heat is transferred by boiling the coolant and vaporizing it from the SFSP into the containment. The source of water for the SFSP in this mode is the spray system (classified as a safety system), which allows, via a branch of the pressure route of any of the three spray pumps, resupplying water from the tank-reservoir of the containment into any of the three SFSP cooling circuits and thus also into any of the three sections of the SFSP. The reservoir in the containment contains about 580 m^3 of coolant.

Another possible method of resupplying the SFSP is to use a pump of the SFSP coolant purification system and using this pump to supply coolant to the reserve coolant tanks for refueling in the SFSP. These pumps are connected only to unsecured power supply sources. The reserve tanks contain about 1,600 m^3 of coolant. Because these tanks are common for both units, the amount of coolant available for resupplying the SFSP in the affected unit in case of refueling in the other unit is limited.

If the reactor loses tightness and the partition between the SFSP and the pool for refueling is removed, there is also the possibility of supplying coolant using any high-pressure or low-pressure pump or pump of the ECCS system into the I.C and from there into the SFSP. Coolant can also be supplied from the hydro-accumulators.

III.1.1.1.5 Heat transfer from the containment to the ultimate heat sink

The projected function of the containment is to limit the potential radiation consequences in case of an accident within the reactor. This function is fulfilled, among other things, by the construction and structure of the containment, which limit leaks outside the containment to very low values even in case of very high overpressure inside the containment. Because the containment contains the entire pressure boundary of the I.C, it is also the last barrier preventing radionuclides released from the fuel or coolant from escaping in case of an accident.

The integrity of the Temelín NPP containments was ensured in the design by the following safety systems:

- Containment insulation system – separating valves automatically closed in case the pressure in the containment increases
- Containment pressure reduction system – spray pumps from the reserve tank with chemical reagents to capture post-accident iodine.
- Post-accident hydrogen liquidation system – passive autocatalytic recombinants, design basis accidents.

Calculated project load in the containment is as follows:

Maximum temperature	150 ^oC
Maximum pressure	0.49 MPa
Pulse power input	10^3 Gy/hr
Tightness	0,1 %/24 hours.

Under normal circumstances, the containment is cooled by systems located inside the containment. In case of an accident, during which the overpressure in the containment exceeds the value of 0.3 kPa, these systems are disconnected and stop working. In addition, the safety valves on the ventilation systems, which ensure underpressure in the containment, are closed. In emergency conditions, the heat from the containment is transferred, and pressure is reduced, by the spray system in combination with other safety systems.

III.1.1.1.5.1 Existing heat transfer means

To transfer heat from the containment, it is possible to use operating systems (classified as non safety related systems or safety related systems) in combination with safety systems, as well as systems classified as safety systems.

<u>Operating systems transferring heat from the containment</u>

Normal operating systems, such as containment air circulation systems, are used for transferring heat from the containment into the ultimate heat sink. Heat is transferred from the containment via heat exchangers, from which heat is transferred by the ESW system. The ESW system passes the heat into the atmosphere in the cooling basins with sprinkler systems. Alternatively, heat can also be transferred by exchangers, from which heat is passed on using cool water.

<u>Safety systems</u>

In case of accidents during which the overpressure in the containment (damaged tightness of the primary circuit or piping of the secondary circuit located in part inside the containment) exceeds 30 kPa, the heat is transferred from the containment by the spray system. The purpose of the spray system is to condensate the vaporized coolant and thus to reduce pressure in the containment and prevent fission products being released into the surrounding environment. This is achieved by injecting cool water containing boric acid and by the subsequent condensation of steam. Heat is transferred from the containment using the spray system thanks to the fact that water drains into the sump in the containment then passes through the ECCS exchanger, from where heat is transferred into the ESW system, and the water is returned to the containment.

<u>Other systems</u>

If it is not possible to transfer heat from the containment into the atmosphere using the ESW system via the spray system, it is possible to transfer part of the heat alternatively – by injecting water in the containment. This can be done using an MCP water fire extinguishing system. This system is capable of dispersing water in the containment with similar effect as the normal containment spray system.

III.1.1.1.5.2 Layout information on the heat transfer means

<u>Operating systems transferring heat from the containment</u>

Under normal operating conditions, ventilation circulation systems of the containment are available to transfer heat from the containment. These systems are classified as safety related systems.

The cooling circulation system for SG rooms transfers heat and removes steam from the air using technology kept in a room within the containment. The aim is to keep the temperature of the air within the required limits. Heat and steam are transferred using surface coolers. Hot air is suctioned from the rooms, and after it is treated (cooled) it is returned to the individual boxes.

The system has 2 x 100% redundancy (3 working + 3 spare ventilators); each ventilator is combined with two coolers in serial layout. One cooler is connected to the ESW system and the other cooler to the cool water mains. The ventilators are connected to a secure power supply for safety systems (DG). The ventilators and coolers are located inside the containment.

The circulation system for cooling the reactor shaft is designed to cool the shaft of the reactor. Air is suctioned from the rooms containing the SG, cooled using surface coolers and fed via piping back to the reactor shaft and then into the rooms with the SG.

The system has 3 x 100% redundancy; i.e. it has three ventilators, from which one is operating and 2 are spare. The coolers are connected to the ESW system and system for cooling water. The ventilators are connected to the secure power supply for safety systems (DG). The ventilators and coolers are located inside the containment.

System for the cooling reactor hall – this provides cooling of the top part of the containment. The air is suctioned from the rooms with SG and after being cooled using a two-stage cooler, it is returned to the reactor hall.

The system has 3 x 100% redundancy; i.e. there are three ventilators, one of which is operating and two spare. The ventilators are connected to a secure power supply for safety systems (DG). The first series of air coolers is connected to the ESW system; the second series is connected to the cooled water distribution system. The ventilators and coolers are located inside the containment.

The system for cooling the cluster drive transfers heat from the cluster drives. The air is suctioned from the rooms in the containment and after cooling, blown back into the rooms with the SG.

The system has 3 x 100% redundancy; i.e. there are three ventilators, one of which is operating and two spare. The coolers are connected to a non-essential service water supply. The ventilators are connected to a secure power supply for safety related systems (DG). The system is located in the reactor hall inside the containment.

The circulation systems, with the ESW cooling the exchangers, can transfer heat from the containment for a practically unlimited period of time.

<u>Safety systems</u>

The spray system in the containment is designed to maintain conditions in the containment in case of an accidents. Pumps suck the coolant from the containment sump via the ECCS exchanger and inject it into the containment with spray jets. The steam condensates on the droplets of water, and thus the pressure in the containment decreases. The water drains into the containment sump from where the heat is transferred into the atmosphere using the ESW system in the ECCS exchanger.

The spray system has 3 x 100% redundancy, including all auxiliary systems (cooling, power supply, control and ventilation), and it comprises three technologically and functionally identical independent subsystems, each of which is capable independently of carrying out the function for which the spray system was designed. Each of the 3 subsystems contains a spray pump, tank with a solution of H_3BO_3, N_2H_4, KOH (to capture iodine vapor), water pump, ECCS exchanger, connecting piping and system of spray jets. The pumps are connected to a secure power supply for safety systems (DG).

The spray system provides an emergency supply of water for the SFSP, if it is necessary.

The reduction of pressure and transfer of heat using the containment spray system is carried out within enclosed circuit via the containment sump.Therefore, heat can be transferred from the containment in this mode for an indefinite period of time.

<u>Other systems</u>

Alternatively, it is possible to transfer the heat from the containment by dispersing water in rooms containing MCP engines. Although this system was designed for extinguishing fire in the MCP engines, it is connected to other rooms in the containment, and therefore, it may be used with similar effect to the standard containment spray system.

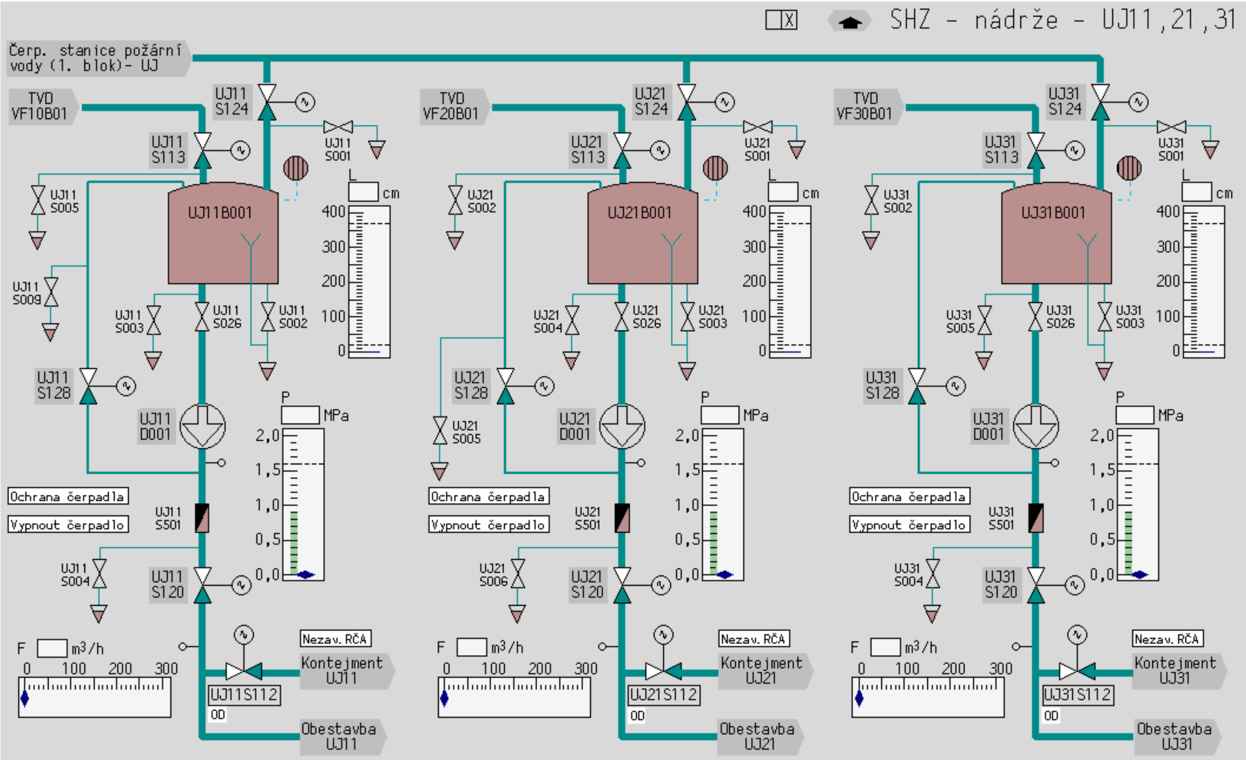


Fig. 33: Fire protection system

The pumps of the fire protection system suction water from tanks containing 3 x 70 m^3 of water and, via jets for extinguishing fire in the MCP engines, inject water into the containment.

The fire protection system has 3 x 100% redundancy, including all its auxiliary systems (cooling, power supply, control and ventilation). It consists of three technologically and functionally identical and independent subsystems. The pumps are connected to a secure power supply for safety systems (DG). The pumps and reserve tanks are located outside the containment.

The volume of the reserve tanks was calculated to be sufficient for extinguishing fire. If the containment is to be sprinkled with water, it is possible to resupply the tanks of the fire protection system from the ESW system. This combination offers an alternative method of injecting and dispersing water in the containment for a practically unlimited time.

III.1.1.1.6 AC power supply

In terms of the power supply systems in the Temelín NPP, in connection with the machine and nuclear systems, the following principle of "defence-in-depth of power supply systems" was applied:

Tab. 24: Levels of defence-in-depth for the power supply part in the NPP

Level DID	Entire NPP	Power supply systems in the NPP	Sturdiness of levels
1	Preventing deviations from normal operation	• Insensitivity to deviations U,f • Stability of energy transfer • Dynamic stability • Off-grid operation	• Related to the sturdiness of the technology, OCS and the construction • Sturdiness of power supply systems (independence, redundancy, diversity) • Protections • Regulation, automatic systems • Quality • Testing of functions • Operating instructions • Trained personnel • Etc.
2	Identification of events and rectification, states and conditions of abnormal operation	• Regulating TS to HC • BES to reserve power supply	
3	Actions (measures) averting development of or gaining control over emergency conditions using design means	• Project (safety) functions of the system of secure power supply: • Safety system (1, 2, 3), • Safety related system (4, 5)	
4	Preventing and mitigating consequences of beyond design basis conditions	• Procedures for coping with SBO • Measures supporting mitigation of consequences of SA • (Function AAC)	
5	Measures for protection in case of a radiation accident	Support for emergency control centres	

Resilience against external and internal events (failures) is a result of the DID structure and sturdiness of individual levels.

Important protection and control systems in the Temelín NPP and high-performance systems designed to fulfill safety functions are connected to redundant secure power supply systems (SPSS). Each unit has 3 independent safety SPSS (1, 2 and 3) and other SPSS, classified as related to safety (SPSS4 and SPSS5). These SPSS provide support safety functions, such as a secure power supply, and take part in control function over electrical appliances.

III.1.1.1.6.1 Off-site power supply

The Temelín nuclear power plant is located in the south of the Czech Republic, consisting of two VVER 1000 MWe units. The plant discharges the generated power output to the electrical grid and receives its power supply for internal consumption from the grid (operating and reserve power supply systems) in accordance with this basic concept, i.e. separately for each unit. The power output is discharged to the 400 kV electrical grid. An auxiliary power supply is secured from the 110 kV electrical grid.

<u>Connections of the plant to external electrical grids</u>

The turbine set of each unit contains a 1111 MVA, 24 kV generator. The power output from the generator is led via a generator switch and unit transformer (1200 MVA, 420/24 kV) using independent simple line into the Kočín 400 kV switchyard, about 3 km from the Temelín NPP. The 400 kV switches for the units of the NPP are located in this switchyard.

The 400 kV switchyard is connected to an external electrical grid by 5 lines, which carry the power output to various remote parts of the Czech Republic (Central Bohemia, West Bohemia, South Bohemia). This ensures geographical diversity of the 400 kV lines. Some of the lines are simple, some are double. The Czech electrical grid as a whole is designed and operated in accordance with the N-1 criterion. The output from the Kočín switchyard is led into the grid; however, it is designed in compliance with the stricter N-2 criterion. These requirements are defined in the PS Code.

The Kočín 400 kV switchyard is an outdoor station with short circuit resilience of 50/125 kA. It contains two systems of connections and an auxiliary connection. The units of the NPP are linked to a part of the switchyard, which was designed to meet higher requirements on reliability and resilience against failures. This part of the switchyard follows a scheme of 4/3 switches per branch; 4 of the 5 400 kV lines connecting the switchyard to the external grid are connected to this part. The 5th 400 kV line is connected here, two 400/110 kV, 250 MVA transformers to supply power to the Kočín switchyard 110 kV and the inductor, which contributes to the control of the apparent power output in this hub.

The Kočín 110 kV switchyard was built mainly as the primary source of auxiliary power supply for the Temelín NPP. In addition, the water pumping station for the Temelín NPP located at the nearby Hněvkovice reservoir on the Vltava River is supplied from Kočín. Furthermore, the 110 kV power grid in the South Bohemia region is supplied from this switchyard. The Kočín 110 kV switchyard has a sturdy and flexible scheme, with 3 systems of connections.

The scheme of the Kočín 400 kV and 110 kV switchyards, as well as their operation, were designed to minimize the possibility that failures would spread from one unit of the NPP to another, or from the units to the power grid and vice versa.

Kočín cooperates in a parallel setup with the Dasný 400/110 kV switchyard, about 30 km away, which also contains two 400/110 kV transformers with the same power output as those in Kočín. The Kočín and Dasný switchyards are connected in parallel setup by two lines at both the 400 kV and 100 kV levels. This setup provides backup for the 400/110 kV switchyards, while maintaining sufficiently hard voltage for an auxiliary power supply of the NPP's house consumption in the Kočín 110 kV switchyard.

The Kočín 110 kV switchyard can, therefore, be supplied from geographically and directionally diversified sources within the electrical grid (400/110 kV switching in Kočín and Dasný, 220/110 kV switching in Tábor), as well as within the 110 kV electricity distribution (HPS Lipno).

The units of the Temelín NPP are capable of working in isolation from the electrical grid – off-grid operation.

The TG control system includes a specific off-grid (island) regulator (proportional frequency regulator), whose primary function is to maintain frequency within the off-grid network. The TG “off-grid” mode is activated when a deviation in the network frequency is detected. For scenarios involving considerable reduction of the load on the TG, there is a system that detects high acceleration of the TG and subsequent pulse sent to the TG hydraulic regulation (i.e. regulation of overrun). The TG being in the “off-grid” mode is also an input information for the main regulation of the unit, which allows coordinating the modes of the reactor, TG and RSC. There is also a specific procedure for “off-grid” operation included in the operating instructions for extraordinary modes of operation.

The frequency range of the off-grid network, which would permit long-term operation of the units of the NPP, is limited by the setup of the frequency relay (if the frequency falls below 47.9 Hz or exceeds 51.5 Hz, the unit is automatically disconnected from the grid and regulated into the own-consumption mode).

The ability to regulate off-grid operation was successfully tested in both units during the commissioning period in 2001 - 2003. The tests proved the high quality of the regulation system controlling the speed of TG revolutions using the off-grid regulator, as well as high quality of the other functions supporting the operation in the off-grid mode. The units of the NPP continued to operate in real off-grid operation when the UCTE network fell apart and split into three isolated units on November 4, 2006.

The ability of the NPP units to operate in the off-grid mode is certified by the certification authority as an auxiliary service for the operators of electrical and distribution grids in the Czech Republic.

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

Since the Temelín NPP was put into operation (the NPP was put into permanent operation in 2002), no failure in the 400 kV and 110 kV grids suggesting unsatisfactory function or reliability of the external power supply system or incorrect reaction of the NPP to failures in the external grid has been recorded.

During the commissioning of the units and the subsequent regular operation, several failures were observed, which activated the electrical protection elements. These events were related mostly to the activation of the units and harmonization of systems. The following disturbances can be reported:

Significant failures in the power supply part of the NPP

Failures recorded while the units were being put into operation and during the test run were analyzed in detail and corresponding corrective measures were defined and implemented. During a big failure in the UCTE network in 11/2006, the NPP units showed high resilience against grid failures.

The reliability of the connections of the NPP with the external electrical grid and resilience of the NPP against failures is based on the following properties:

a. Unit layout of output discharge into the external electrical grid. This limits the transfer and spread of failures between the units. In combination with a strong layout of the Kočín switchyard (4/3 switches per branch, sectional division of connections, selective protection system) it also limits the transfer of failures between units and the electrical grid. The discharge of the output of the NPP into the grid is designed to comply with the N-2 reliability criterion.

b. The functional and physical independence of the 400 kV output discharge system (i.e. functional operating supply for house consumption) and the auxiliary power supply system for house consumption (110 kV). The possibility to get an auxiliary power supply from geographically and functional diverse sources.

c. The reactions of the units to failures and transitional processes in the external electrical grid are controlled by a set of regulations, automatized reactions and protections. These functions are coordinated to ensure mutual selectivity and to make sure that, if necessary, the units are able to retreat in a controlled manner down the levels of defence-in-depth.

d. Static stability of the power discharge into the electrical grid. Under normal circumstances, the units of the NPP are included in the system for automatic secondary voltage and idle output regulation. This system ensures voltage stability in the Kočín pilot hub and controls the position of lower limits for generators, depending on the external impedance of the electrical grid.

e. The stability of the turbine set during short circuits within the output discharge system (quick basic and backup protections that break the circuit, effective regulation of the turbine and voltage on the generator, quick control of valves on the turbine). The stability was analyzed using a dynamic model of the electrical grid. While the basic and backup protections function (up to 100 ms), the turbine sets are naturally stable. In case short circuits last longer (automatic reaction systems if the 400 kV switch fails), stability is enhanced by a function providing quick control of the turbine valves.

f. The capacity of the units to work in an "off-grid" setup usually combined with large jumps in frequency and voltage (support of network stability in case of systematic failures). The units can work at full power only within a range of 49÷50.5 Hz. The time-limited and output-limited functioning of the units is possible within a range of 47.5÷51 Hz. The units contain network frequency protectors, which in the first-level protection (±200 mHz) switch regulation of the output of the units to "off-grid" regulation. The unit thus reacts to its own output in a way that helps stabilize the ratio of U and f in the "off-grid" network. If the situation does not stabilize and the frequency deviation continues to grow and exceeds the limits (47.9 Hz for 1 s or 51.5 Hz for 10 s), the unit uses second-level frequency protection to disconnect from the external network and regulate house consumption.

   The reactor protection system measures the output of the MCP and is, therefore, sensitive to the frequency and rate of decrease of frequency in the network. In case of sudden frequency decrease (output imbalance), it is preferred to shut down the reactor by the protection system rather than to regulate the turbine set to house consumption. In this case, the power for house consumption is either supplied from a backup source from the 110 kV network (automatic replacement), or secure power supply systems and emergency DG are activated.

g. The unit's main switchboards contain automatic functions that switch from operating to auxiliary power supply (110 kV). There are quick and backup channels to compensate for loss of voltage. In case the operating power supply for house consumption is lost (e.g. after the activation of generator protections, the unit's switchboards protections and protections on other elements of output discharge or in case of unsuccessful regulation of the turbine set), the switchboards are switched to auxiliary power supply. If the substitution is unsuccessful, secure power supply systems are activated and the system relies on emergency sources (DG, accumulators).

h. Regulation and automatic elements and protections are connected to the power supply from batteries. Their functioning is, therefore, not affected by decreased voltage in the network caused by failures. The principle of electromagnetic compatibility, which ensures the functioning of the systems in a given electromagnetic environment and disturbances, is applied throughout the Temelín NPP project.

i. The NPP units operate in cooperation with control centre, which oversees the electrical grid. The electrical grid operator is familiar with the properties and operational limits of the Temelín NPP – information fixed in the PS Code. Regular maintenance of the external facilities (switchyards, lines, 400/110 kV transformers) and of the systems of the NPP is carried out in an agreed manner. In case of an emergency in the electrical grid (breakdown of the network, SBO of the NPP etc.), the operator's first priority is to reestablish the power supply for house consumption of the NPP.

III.1.1.1.6.2 Power distribution inside the plant

Main cable routings and power distribution switchboards

The power supply for house consumption of electrical appliances is provided by several switchyards, supply systems and sources with backup (based on substitution and redundant principles). Thus, the consequences of failures within these systems for the operation of the units and reactors within the units are limited.

Electrical appliances are divided into groups according to their importance and depending on whether or not they are supplied from sources and networks within the corresponding category of power supply safety. The importance of an appliance is based on the safety function of the appliance and permissible time of no power supply. The safety function of an appliance is based on a classification according to IEAE standards: Safety systems (SS), safety related systems (SRS) and non safety related systems (NSRS). Own consumption of each unit of the Temelín NPP can be supplied by:

- **Operating sources**, i.e. house transformers with voltage regulation (supplied from the TG 1000MW and/or from the 400 kV grid). The operating sources are unit-specific.
- **Auxiliary sources**, i.e. reserve transformers with voltage regulation (supplied from the 110 kV grid). The reserve transformers are separate for each unit, but can also substitute transformers in a neighboring unit. The auxiliary transformers are able to shut down one unit if the operating power supply is lost, even while supplying power to the other unit.
- **Emergency sources**, which supply power to the secure power supply systems (SPSS). Emergency sources are diesel generators, batteries and generators of uninterrupted power supply (rectifiers, alternators). These are installed in the Temelín NPP compound; their parameters comply with the requirements based on the necessary supply loads and their operability does not depend on the state of operating and auxiliary sources, or on the external grid. Each unit in the Temelín NPP has 3 redundant SPSS, classified as safety systems (each of which is an auxiliary system for its own division of safety systems) and two SPSS that supply safety related systems and non safety related systems.

Appliances that are not important for nuclear safety (SRS, systems ensuring operation of the unit, production of electrical power) are connected to operating power sources. In case the operating power supply is lost (house transformers), these systems are automatically switched to auxiliary power supplies (reserve transformers). Each unit contains four 6 kV main unit switchboards, each consisting of two sections. Section "a" provides power supply to the MCP and is fitted with BES quick control logic to prevent the activation of reactor protection in case of operating power supply failure. Quick BES has a backup – conventional BES. Section "b", which supplies power mainly to appliances of the secondary circuit, contains conventional BES.

Appliances important for nuclear safety (SS and SRS) are connected to secure power supply systems (SPSS). The SPSS consist of secure power supply grids and emergency power sources. The SPSS are normally connected to operating or auxiliary power sources. In case this power supply is lost, the SPSS are disconnected from the standard supply network and connected to power supply from emergency sources. Uninterrupted power supply for sensitive appliances is provided by batteries.

The disconnection of SPSS and launching of DG are initiated at the moment power supply is lost (U < 0.25 Un for 2s, with backup logic based on the state of the switches in the 6 kV unit switchboard for operating and auxiliary power supply). Frequency fluctuations are managed using network frequency protection, which evaluates frequency decrease in the 400 kV grid. Project analyses and tests have confirmed the selectivity of this setup with respect to the BES function from operating to auxiliary power sources.

After this initialization, DG are automatically launched and put under load by the given ELS program. In accordance with the safety requirements, the DG are ready for load 10s after the launch instruction. The functionality of DG and their automatic load-bearing capacities are verified periodically by testing.

Layout, location, and physical protection against internal and external hazards

Unit, branch and reserve transformers are located in front of the machine hall. These locations are physically and electrically separated and protected from fire individually.

Operating sources for house consumption (2 house transformers, each with an output of 63/31.5/31.5 MVA) are supplied from the main generator, which supplies the unit's main 6 kV switchboards, located in the switching room next to the machine room. This building also contains 6/0.4 kV reduction transformers and 0.4 kV distributors for supplying the machine hall and secondary circuit.

Reserve sources for house consumption (2 reserve transformers, each with an output of 63/31.5/31.5 MVA) are supplied from the 110 kV grid. Transformers in each of the units can be used as backup for the other unit using 6 kV connections. These transformers provide a reserve power supply for the unit's 6 kV switchyards.

The unit's 6 kV switchboards also supply 6 kV motors (e.g. MCP), 6 kV switchboards within the safety supply system and 6/0.4 kV transformers for the appliances in the reactor buiding. These distribution systems are located in the hall surrounding the reactor chamber.

The unit's 6 kV switchboards also supply 6 kV switchboards, 6 kV drives and 0.4 kV switchboards located in external buildings (pumping station, compressor station, building containing auxiliary active systems etc.).

III.1.1.1.6.3 Main ordinary on-site source for backup power supply

On-site sources that serve as first backup if off-site power is lost

Safety systems (SS) in each unit of the Temelín NPP are organized into 3 divisions of safety systems (3 x 100%). In accordance with this concept, each division (referred to as 1, 2, 3) includes an SPSS (also referred to as 1, 2, 3), which serves as a backup safety power supply for appliances within the corresponding division.

To ensure the necessary degree of redundancy, these SPSS are independent and separate (constructional and fire-protection separation) with a separate power supply and a link to the control system. SPSS 1, 2 and 3 are seismically resilient. Each SPSS has its own emergency sources (DG, accumulators), as well as power mains. SPSS 1, 2 and 3 also supply systems with a lower classification in terms of safety (SRS – safety related systems, or NSRS – non safety related systems), which require a high level of reliability and redundancy. However, these systems cannot compromise the ability of the safety systems to carry out safety functions.

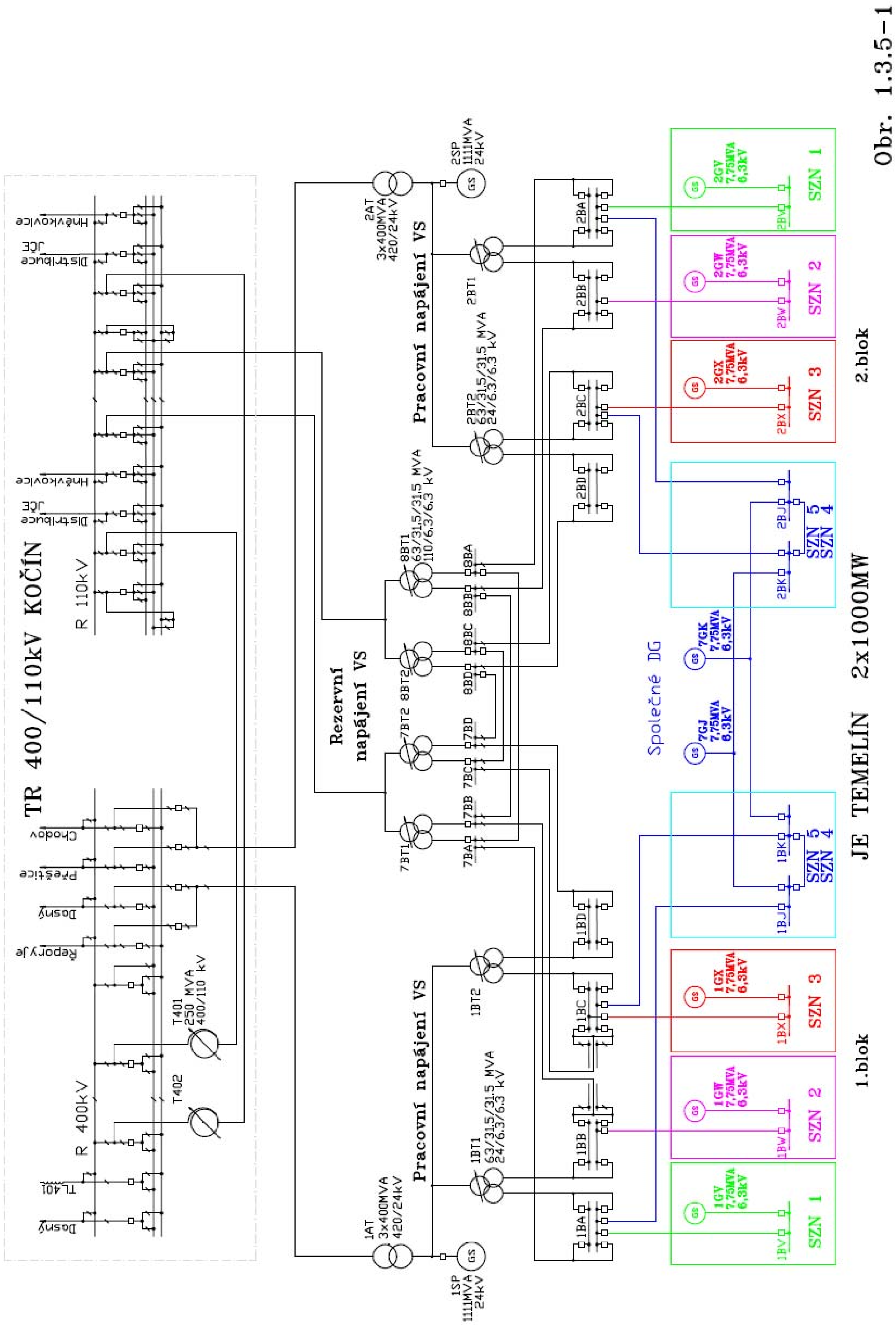


Fig. 34: Basic scheme of house consumption in the NPP Temelín

Each SPSS (1, 2 and 3) consists of the following main components:

(1) Emergency DG 6.3kV, 6.3 MW. The diesel generators (GV, GW, GX) have own diesel fuel tanks with fuel sufficient for at least 48 hours under full load without refueling (in fact, for longer, because the real load is lower). Fuel can be supplied from tanks within the diesel fuel management system.

(2) 6 kV switchboard of a secure power supply system.

(3) 0.4 kV switchboard and 6/0.4 kV reducing transformers.

(4) Rectifiers, batteries, alternators supplying sensitive appliances, which require a stable and uninterrupted power supply.

SS divisions 1, 2 and 3 and their SPSS 1, 2 and 3 are fully redundant (concept 3 x 100%). Thanks to the principle of independence and separation, a simple malfunction in one of the SPSS 1, 2 and 3 will not compromise the functionality of the remaining two divisions.

The DG are an emergency source of power for appliances that can handle the interruption of the power supply for a certain period of time (from tens of seconds to minutes). The DG are started automatically when the power supply for the 6 kV switchboard, providing a secure power supply for the corresponding SPSS, is interrupted. At the same time, this switchboard is disconnected from the ordinary power supply by two serially connected sequential switches. The load on the DG and functioning of the SPSS and appliances are controlled by the emergency load sequencer (ELS) with the highest priority according to fixed programs without the need for intervention from the staff. The sequencer will also protect the DG from overload from potential incorrect actions by the operator.

To confirm the permanent readiness of the safety systems, functional tests of the DG and SPSS are carried out even when the unit is in operation (the DG are started and take over the load by the ELS after intentionally switching off the sequential switches to simulate a real loss of power supply).

Two more SPSS (4 and 5) were built in each unit to provide a power supply for safety related systems (SRS) and for those non safety related systems (NSRS), which ensure the general safety of persons and expensive systems, such as the turbine set. These SPSS were designed as two sub-systems (4.1, 4.2; 5.1, 5.2), providing mutual backup according to the 100 + 100% principle.

The main emergency power source for these SPSS is a pair of diesel generators, GJ and GK (6.3 kV, 6.3 MW each), common for both units of the NPP. These common DG, and most supplied SPSS, are not seismically resilient. One working DG is sufficient for supplying power to these SPSS. This is also the solution for the ELS, which provide a gradual increase of the load for these DG. The fuel tanks contain enough fuel for each common DG to run for about 12 hours under 100% load (providing a power supply for all SPSS in both units). In a realistic scenario, both DG work with a lower load and therefore, their operating time will be longer. Each of the sub-systems has its own accumulators, rectifier and alternator.

In the procedures for dealing with SBO (station blackout), these DG and SPSS are described as on-site reserve AAC. Common DG are partially diverse (at least in terms of their locations and their positions in the power diagram, as well as by the controlling sequencers) from the DG at SPSS 1, 2 and 3. Common DG can be connected to the switchboards of SPSS 1 and 2 by manipulating the existing 6 kV mains.

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

SPSS 1, 2 and 3 (including DG) consist of seismically resilient elements located in seismically resilient buildings.

The buildings of the DG stations for SPSS 1, 2 and 3 are sturdy reinforced concrete objects protected against simultaneous failures by their chessboard layout. The DG system for SPSS 1 is located on the other side of the reactor chamber than the DG for SPSS 2 and 3. DG 1 for Unit 2 is located between DG 2 and 3 for Unit 1 (these DG systems are between the NPP units). DG 2 and 3 for unit NPP 2 are located on the other side of the reactor chamber of unit NPP 2.

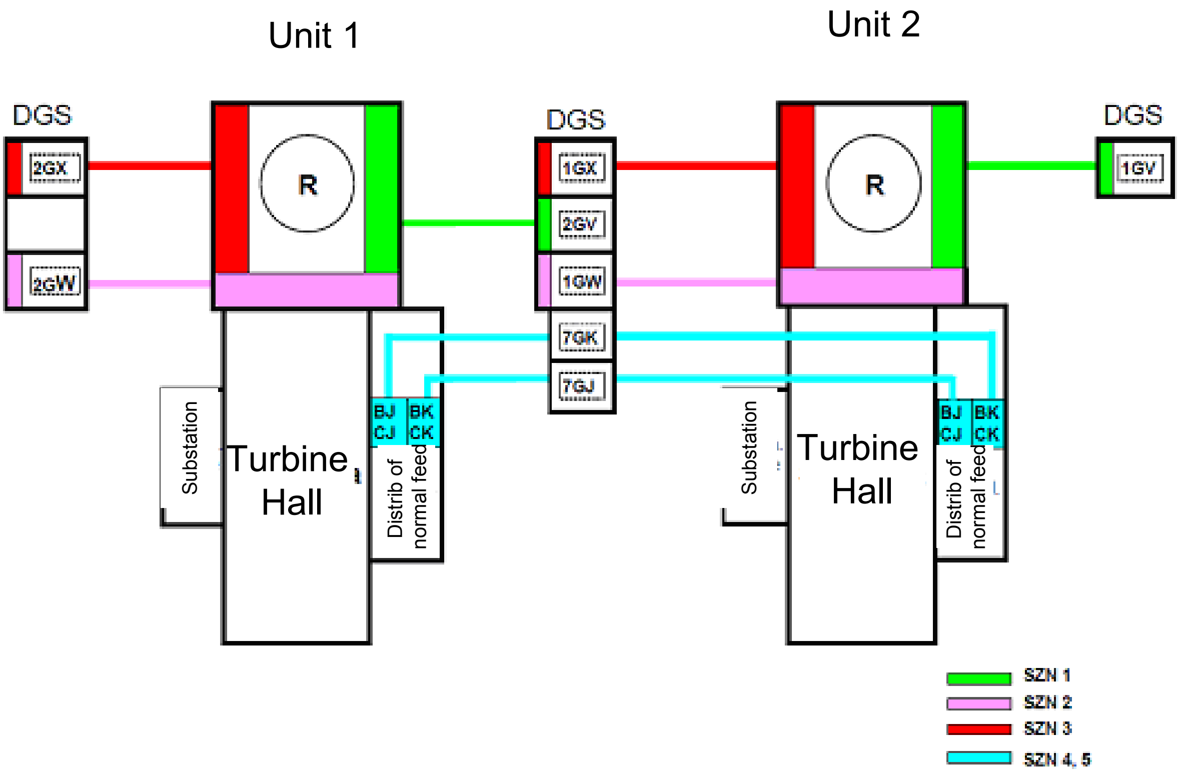


Fig. 35: Layout and positions of power sources and SPSS

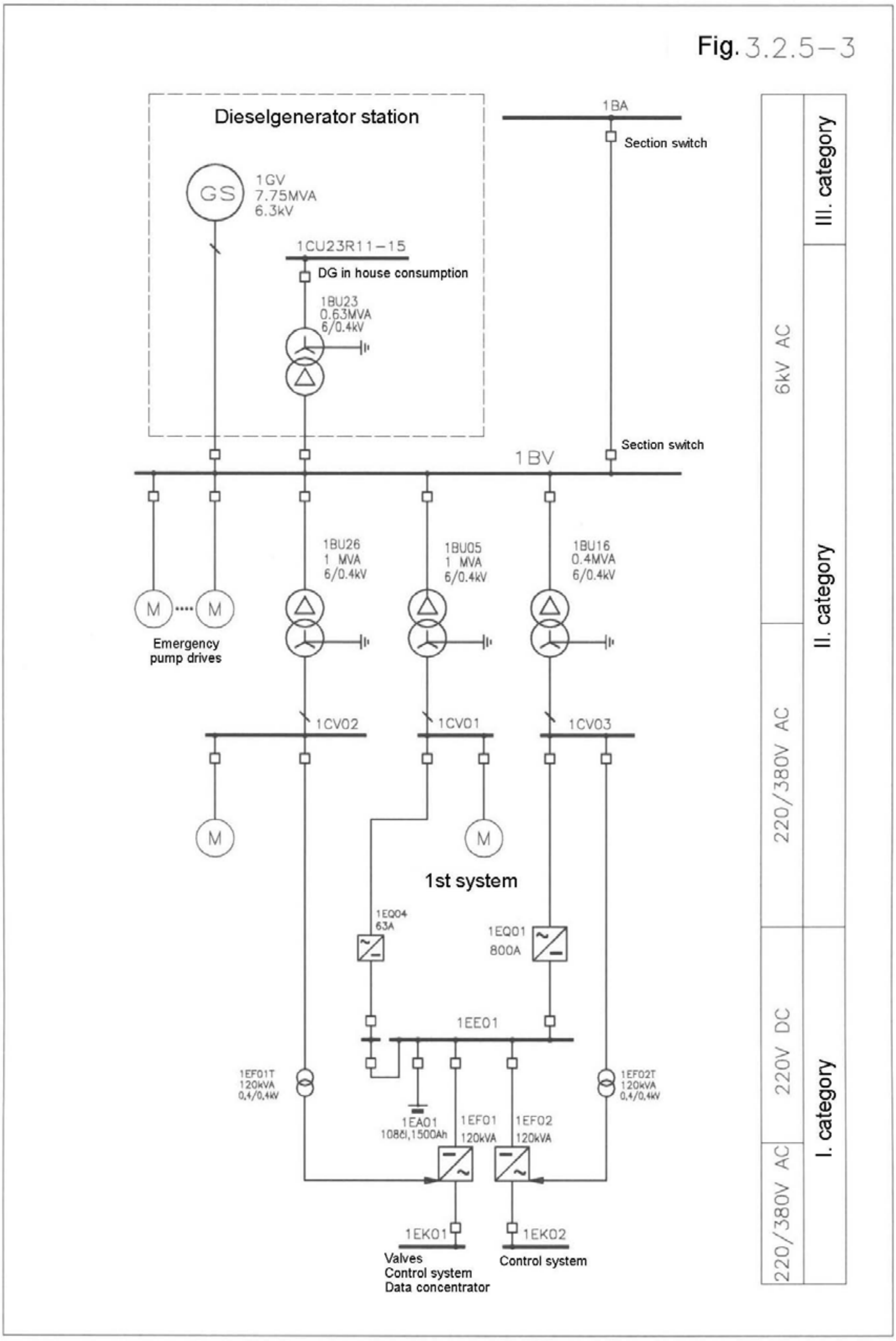


Fig. 36: Power sources for house consumption

Power facilities of each of the SPSS 1, 2 and 3 (switchboards of 6 kV, 0.4 kV, batteries, etc.) are located on three different sides around the chamber room, which protects them from external risks.

The cable routes for SPSS 1, 2 and 3 are independent. This guarantees functional, spatial and fire-protection independence (90 minutes) of these SPSS and the corresponding divisions of the safety systems.

All auxiliary systems of a DG's engine and generator (fuel intake, lubricating oil, internal cooling circuit, air, starting air) are autonomous and, while the DG is in operation, independent of the external power supply. Each DG has own grid system and own consumption system, including a battery. Systems that could be negatively influenced by the long-term operation of the DG (e.g. clogging of the oil system filter) have redundant subsystems, so that one can be shut down while the DG is in operation to carry out maintenance and thus prevent failure of the DG due to a loss of the auxiliary systems. The quality of diesel fuel is checked monthly and kept in accordance with the corresponding requirements.

SPSS 4 and SPSS 5, as well as their common DG and cable routes, are not designed as a whole to be seismically resilient. One exception is SPSS 4.1, located in the hall around the reactor building. However, redundant switchboards and appliances are supplied using cables leading along different routes, in order to increase reliability and ensure functionality in case of fire.

Common DG are located in separate rooms in the common DG building located in protected area between reactor units Temelín NPP 1 and Temelín NPP 2. The building containing the DG station provides strong protection against most external influences (adverse weather conditions, etc.). The majority of the distribution systems of SPSS 4 and SPSS 5 are located in the buildings containing switchboards (building 500, next to the turbine hall).

Besides tanks located near the DG (with a sufficient diesel fuel reserve for DG 1, 2 and 3 for at least 48 hours), the compound of the NPP also contains a diesel fuel management system with 4 tanks, each of which contains 1,000 m^3. If necessary, it is possible to resupply fuel into the DG tanks using mobile means.

Time constraints for availability of these sources

Loss of power is evaluated by two systems. Each system identically generates two signals to evaluate loss of voltage. If the value of any combination of voltages from the three-phase set drops below 25% of the nominal value, the ELS (emergency load sequencer) will register one of the signals. In case a counter-electromotive force appears together with the voltage drop, the signal for launching the ELS will be delayed. This delay represents the time during which the protection system must be able to switch off the failure (type "asymmetric short-circuit") in the supply grid. To increase reliability, the signal is created under "negative logic", i.e. the ELS is activated if the signal has value of logical zero (i.e. no voltage).

The generated signal "loss of power supply section 6 kV category II" lasting 30 seconds leads to the following:

- Activation of memory "loss of power" (LOOP).
- Signaling "Start ELS" – in the CR (unit control room) and ECR (emergency control room) while the memory "loss of power" is active.
- Both switches of the sectional connection are switched off and consequently, the power supply for the affected division of safety systems is separated from the operating power supply.
- Selected loads are switched off.

- Interlocks are generated to block both manual and automatic control of the operational controlers of selected loads.
- DG is started (delay of 0.2 s).

After the DG has been successfully started and the required parameters are achieved, and once at least one 6 kv swithyard section switch has been switched off, the automatic system switches on the DG. Once the DG is switched on, the program for the distribution of the load is also started (second "zero" of the ELS program). Subsequently, selected appliances are gradually switched. The ELS program will be changed depending on the technological conditions in the unit. If $T_{I.C} < 70^oC$ load on the DG is controlled by the ELS-C program. For $T_{I.C} > 70\ ^oC$, it is controlled by the ELS-H program. These programs essentially differ by the list of appliances that will be switched on and by the number of stages.

Termination of the "ELS-H" program (after 30 seconds) or "ELS-C" (after 20 seconds) generates "End of Program ELS" for the CR and ECR signal. Once the ELS program is terminated, interlocks generated in operational controlers are no longer in place, however, certain loads remain blocked for manual switching depending on the output of the DG.

III.1.1.1.6.4 Diverse permanently installed on-site sources for backup power supply

Location, physical protection and time constraints

The ability of the Temelín NPP to cope with a complete loss of off-site power supply (station blackout – SBO) and recover has been analyzed. The basic methodology for an SBO is based on US NRC RG 1.155. The aim of the SBO analysis was to demonstrate that each unit is capable of handling an accident beyond design basis resulting in an SBO mode lasting for a certain period of time and subsequently recovering, maintaining safety functions all along.

The following SBO definition was used:

- The entire NPP (i.e. both 1,000 MW units) lost operating and reserve power supply for house consumption from the external grid (400 kV and 110 kV).
- The 1,000 MW alternator in one of the units (e.g. A) was not regulated and failed, and the DG in SPSS 1, 2, 3 also failed in all three divisions of the safety systems. Only sources supplied from batteries remain in operation.
- The safety of the other unit (B) is ensured by at least one functioning division of the safety systems.
- No design accident or failure was registered immediately before or after the SBO; the following in particular are excluded: Seismicity, fire, floods. All systems in the power plant, besides those systems that caused the loss of power supply for house consumption, continue to function or are able to function.

In order to cope with the SBO, an auxiliary source of alternating current (AAC) was connected. The analysis revealed the following main limits and requirements in case of an SBO:

- Using methodology from UR NRC RG 1.155, the maximum period for which the Temelín NPP must be able to cope with an SBO is 8 hours.
- It is necessary to provide a power supply for the ESW system and an emergency supply for the SG.
- In the period between 2 and 6 hours after the SBO event, cooling of the spent fuel pool must be ensured (depending on the content).

When these measures are implemented, the unit is capable of remaining in hot state, in terms of water reserves in the system for emergency water supply for the SG, for about 56 hours.

If a unit needs to be cooled to a cold state, the following steps are necessary:

- Within about 48 hours after the SBO event, activate the emergency supply systems to increase the concentration of boric acid in the I.C and transfer heat from the I.C.
- Before launching the process of cooling the unit to a cold state, the containment must be isolated, otherwise during pressure reduction in the I.C, and possible implementation of the "feed & bleed" method, the coolant will be released into the containment. The safety valves of the system for emergency bleeding of the I.C must be activated, as well as the valves on routes between the primary circuit and hydro accumulators. The spray system for the reduction of pressure in the containment must be activated.

All these measures require maintaining the functionality of all necessary I&C (oversight and control system) and ensuring the functionality of the corresponding auxiliary systems:

- Ventilation that provides cooling of the rooms containing systems for emergency water supply for the SG and for the I.C, electrical rooms and I&C. According to analyses, failed ventilation would lead to a critical temperature increase in some I&C rooms more than 60 minutes after an SBO event.
- Power supply systems (Categories I and II).

Mechanical and nuclear requirements imply the need for restoring the functionality of at least one division of safety systems 1, 2 and 3. By summarizing the output requirements, the following value of the necessary output was calculated:

- To keep a unit in safe hot state:
  Pp = 2.5 MW (max. engine Pn = 800 kW) (supply restored within 35 min.)
- To keep a unit in safe hot state with subsequent transition into a cold state: Pp = 2.5 MW (within 35 min.) and subsequently additional 1.2 MW (max. engine Pn = 800 kW)

Above described philosophy implemented to cope with SBO was confirmed as acceptable during the IAEA design review missions focused on VVER 1000 safety issues resolution.

Diverse sources that can be used for the same tasks as the main backup sources

Based on the conclusions from the above analyses, emergency operating rules were created for the Temelín NPP that deal with SBO-type emergency events with respect to the available power sources and the current state of the equipment. After carrying out activities ensuring power supply for the safety systems, the rules define procedures for various possible states of the electrical diagram in both units. The method to be employed depends on the state of both units of the Temelín NPP and on the state of the off-site power grid. The following on-site AAC sources are considered:

1. 6 kV switchboards in the neighboring unit, which has already adapted operation for house consumption after losing the power supply from the 400 kV and 110 kV power grids. Connection can be established via a 6 kV backup power supply route. Another possibility is to create a connection via the SPSS 5 switchboards.
2. Common DG (7GJ, 7GK) with an output of 6.3 MW. It is clear that in terms of output, one of these DG is sufficient to cope with a postulated SBO within one

unit. Its output is probably sufficient to cope with an SBO in both units of the Temelín NPP. The advantage of this source is its readiness and availability for dealing with an SBO, good protection from the weather and complete independence from the 400 kV and 110 kV off-site power grids.

Other power sources that are planned and kept ready

Depending on the states of the units of the Temelín NPP and on the condition of the off-site power grid, the following AAC sources can be employed to deal with an SBO:

1. 400 kV and 110 kV off-site power grids. The 400 kV and 110 kV off-site power grids can be used if their functionality and connection with the Temelín NPP has not been disrupted. In order to restore voltage in off-site grids, the owner of the power grid (ČEPS a.s.) has internal instructions, of which the number one priority is restoring the power supply for house consumption of the Temelín NPP.
2. The HPS Lipno (output 2 x 60 MW) can be connected to the backup power supply system of the Temelín NPP using 110 kV connections, under the condition that these external lines and 110 kV switchyards on the route are functional. The Lipno HPSwas defined by SBO analyses as the main off-site AAC source, and this functionality was verified by practical tests. The Lipno hydro power plant (output 2 x 60 MW) can be launched “from the dark”. A test proved that this power plant is capable of providing a power supply within 30 minutes. The test included verification of the organizational measures necessary for coping with an SBO, the functionality of the TSPP (technical state of physical protection), the functionality of means of communication, as well as the roles and procedures of key persons in case of an SBO. A condition for using the HPS Lipno (about 60 km from the Temelín NPP) is the functionality of the 110 kV switchyards and cables along the route.
3. small HPS Hněvkovice– source of small output (from 2 x 2.2 MW to 2 x 4.8 MW, depending on water flow). Power can be led to the Temelín NPP via the Kočín 110 kV switchyard, using the 110 kV route of reserve power supply.

Other on-site power supply sources are also available for the Temelín NPP, which are not designed to supply safety systems in case of an SBO; therefore, the possibility of using them for an SBO has not been tested:

- DG supplying oil pumps of the turbine (output 200 kW).
- DG for the data centre (output 1 MW).

III.1.1.1.7 Batteries for DC power supply

Each SPSS contains sources and mains, which ensure uninterrupted power supply for sensitive appliances. These emergency sources are 220 V lead batteries. All SPSS were inspected in connection with events in the Forsmark NPP. Consequently, the settings and coordination of the protection and monitoring systems were adjusted and now ensure resiliency against malfunctions and transitory processes in the AC power grid.

Under ordinary operation, loads are planned and batteries are recharged from normal power sources using rectifiers. In case the operating and reserve sources fail, power is supplied from emergency diesel generators.

The designed rectifiers can recharge batteries in less than 8 hours.

III.1.1.1.7.1 Description of separate battery banks

The systems installed for SPSS 1, 2 and 3 (classified as SS) consist of a thyristor rectifier (220 V, 800 A), accumulator batteries (220 V, 1600 Ah), and two transistor alternators (220/380 V AC, 170 kVA). These systems provide power for the key control, monitoring and protection systems and valves within their respective divisions of safety systems. Also, emergency lighting within the corresponding safety divisions is an important appliance (classified as NSRS).

SPSS 4 contains two subsystems (4.1, 4.2), supplied from SPSS 6 and therefore also from the common DG. Each of these subsystems contains a thyristor rectifier (220 V, 1000 A), accumulator battery (220 V, 2000 Ah), and an alternator (220/380V AC, 170 kVA). These systems are classified as SRS (safety related system) or NSRS (non safety related system). Subsystems 4.1 and 4.2 are redundant (100% + 100%); the appliances have connections from both subsystems.

SPSS 5 contains two subsystems (5.1, 5.2), which can be also supplied from a common DG. Each subsystem contains 2 thyristor rectifiers (220 V, 800 A), accumulator battery (220 V, 2400 Ah), and an alternator (220/380V AC, 170 kVA). These systems are classified as SRS (safety related system) and they supply the appliances of the control system, classified as SRS or NSRS and safety drives for the turbine sets. Subsystems 5.1 and 5.2 are redundant (100% + 100%); most appliances have connections from both subsystems.

Other safety relevant battery systems are in the diesel generator stations. They consist of rectifiers and 24 V batteries. They are supplied from the mains of house consumption from their respective DG. They supply systems and protection of the DG; the discharge period under this load exceeds 8 hours and they are in the same class as DG (i.e. DG for SPSS 1, 2 and 3 as SS, common DG as SRS).

Moreover, the units of the Temelín NPP contain other battery systems. Batteries within the system for the driving reactor's control clusters (110 V, 1200 Ah) stabilize this system during short-term voltage fluctuations, which may occur in the power grid or in the grid for house consumption. Two 24 V, 600 Ah batteries provide uninterrupted power supply for the system monitoring the fall of the control rods to the low position.

The parameters of the batteries were specified on the basis of the requirement of IAEA 50-SG- 07: 1982, i.e. discharge period at least 30 minutes. The batteries were designed and delivered according to this requirement. Subsequently, during the design development, the load on the battery systems decreased, particularly as a consequence of a change of the control system for the system supplied by Westinghouse. As a result, discharge periods were prolonged, but also very unequal.

Tab. 25: Discharge times of accumulator batteries in design modes

SPSS	**Battery**	**Type of battery**	**Discharge time [minutes]** 2) 3)	
		Lead station batteries	LOOP without failure	LOOP + failure 1)
1,2,3	EA01,02,03	108 cells, Vb2415, 1600 Ah	**> 110**	**N/A**
4.1, 4.2	EA04,05	108 cells, Vb2420, 2000 Ah	**500**	**240**
5.1, 5.2	EA51,52	2x108 cells, Vb2412, 2400 Ah	**200**	**95**
SOR 24 V	EA21,22	12 cells, Vb2312, 600 Ah	**> 600**	**450**
SOR 110 V	EA09	54 cells, Vb2412, 1200 Ah	**115**	**N/A**

1) Failure of the battery in one of the subsystems. The functioning battery then carries the entire load.

2) The specified discharge period includes inspection of the supply voltage by loading the battery (stable and peak load), as well as inspection of the battery in terms of used capacity.

[3] The discharge time was specified using the worst time profile of load. The capacity of the battery is reduced to 71% C_{10}, which respects the impact of aging (0.8) and the impact of the minimum temperature in the accumulator room (0.9).

Below are discharge times during an SBO beyond the design project. There are two variants – without a disconnecting load and with a disconnecting load (disconnecting less important parts). With controlled disconnecting, the discharge times of the accumulators are much longer. Controlled disconnecting is included in the manuals for the TSC (technical support centre).

The batteries for the safety systems were inspected in the following alternative way:

a. Partial reduction of the load (after 30 minutes disconnecting 25% of the load).

b. Without reduction of the capacity of the battery (100% C_{10}).

The battery is capable of supplying the system under these conditions for over **4 hours**.

Tab. 26: Discharge times for accumulators during an SBO beyond the design project

SPSS	Battery	Type of battery	Discharge time [minutes] [1) 2) 3)]
1,2,3	EA01,02,03	108 cells, Vb2415, 1600Ah	**>130 / 260** [4)]
4.1, 4.2	EA04,05	108 cells, Vb2420, 2000Ah	**>540**
5.1, 5.2	EA51,52	2x108 cells, Vb2412, 2400Ah	**>200**
SOR 24V	EA21,22	12 cells, Vb2312, 600Ah	**>600**
SOR 110V	EA09	54 cells, Vb2412, 1200Ah	See design modes

f) No failures; all batteries are functional and supply power.

g) Specified discharge time includes inspection of the supply voltage from the battery (stable and peak loads), as well as inspection of the battery in terms of used capacity.

[3] Discharge time was determined using the worst time profile of load. The capacity of the battery is reduced to 81% of C_{10}, which reflects the impact of aging (0.8) and the impact of the minimum temperature in the accumulator room (0.9).

4)Without disconnecting/with disconnecting load

The analysis implies that accumulators are not critical for coping with an SBO beyond the design project because:

a) They can be connected to an AAC source that will support the load and recharge the accumulators via rectifiers.

b) The discharge times of the accumulators are longer than the period during which the supply from the AAC source must be recovered. The same applies to the lifetime of the accumulators.

III.1.1.1.7.2 Consumers served by each battery bank

A direct current power supply is important for the I&C (oversight and control system) systems and for supplying the systems necessary for carrying out the required safety activities, i.e. starting the DG and recovering the power supply, isolating routes for discharging coolant from the I.C and other routes, regulation of pressure in the SG and I.C and isolation of the containment.

Key loads supplied from accumulator batteries of safety systems include:

- I&C systems within the safety systems (PRPS, PAMS, PACHMS)
- Station releasing steam into the atmosphere
- Valves of emergency water supply for the SG
- Safety valves of the SG
- Valves isolating the discharge routes of the I.C
- Separating valves on hydro accumulators
- Separating valves on the bleeding valve of pressuriser

III.1.1.1.7.3 Physical location and separation of battery banks

Battery systems for SPSS 1, 2 and 3 and their rectifiers and alternators are located in the hall surrounding the reactor building. These are seismically resilient systems located in seismically resilient rooms separated by division.

Battery subsystem 4.1 is located in the hall surrounding the reactor chamber, and it is seismically resilient. Battery subsystem 4.2 is located in the building with switchboards (building 500 next to the machine room), and therefore, it is not seismically resilient.

Battery systems SPSS 5 are located in the building with switchboards (building 500 next to the machine room), and therefore they are not seismically resilient.

III.1.1.1.7.4 Alternative possibilities for recharging each battery bank

Accumulator batteries SPSS 1, 2 and 3 can, if necessary, be recharged using an auxiliary rectifier (63 A), which is connected to the ordinary supply power grid. This option can be used in case the main rectifier 800 A is not functional.

In an SBO mode, the batteries can be recharged using available alternative AAC sources.

III.1.1.2 Significant differences between units

Both units of the Temelín NPP are of the same type and there are no significant safety differences between them.

III.1.1.3 Scope and main results of Probabilistic Safety Assessments

PSA (Probabilistic Safety Assessments) for Units 1 and 2 were carried out in the 1993 – 1996 period. The analyses for the Temelín NPP included a level 1 analysis (level 1 PSA) for operation under power low power) operation including shutdown, risk of external events, risk of seismic events and subsequently also a Level 2 PSA.

The original probabilistic models were updated in 2002-2003 to capture the actual state of the design as of the date the units were commissioned after all safety improvements were implemented. The updating of the models included an analysis of fire risks, risk of floods and update of the Level 2 PSA models. The Level 2 PSA currently includes only power operation.

In 2010, the inspection of the reliability of data specific for the location of the Temelín NPP was updated, and new specific data replaced the generic data used until then. As a result, CDF (core damage frequency) improved slightly.

The PSA for the Temelín NPP was subject to the inspection missions of IPERS IAEA in 1995 (Level 1 PSA, internal initiating event) and in 1996 (fire, floods, external events and Level 2 PSA). The following IPSART mission took place in 2003 after this analysis was updated. Also, independent evaluation of the PSA study, requested by the SNSA, was carried out by the Austrian company ENCONET Consulting in 2005. A regulatory inspection of PSA are carried out by SÚJB every year.

Probabilistic models of the PSA are updated regularly as a part of the Living PSA concept – a concept accepted by the operator in response to the SÚJB requirement for regular updating of PSA models so that their results reflect the current state of the NPP and thus meet the basic requirement on their usability for risk-informed applications.

The following graph contains the history of Level 1 PSA (CDF) results for an internal initiating event for:

- o At power operation of the Temelín NPP under power (Mode 1) and
- o low power) operation/shutdown (Modes 2 through 6)

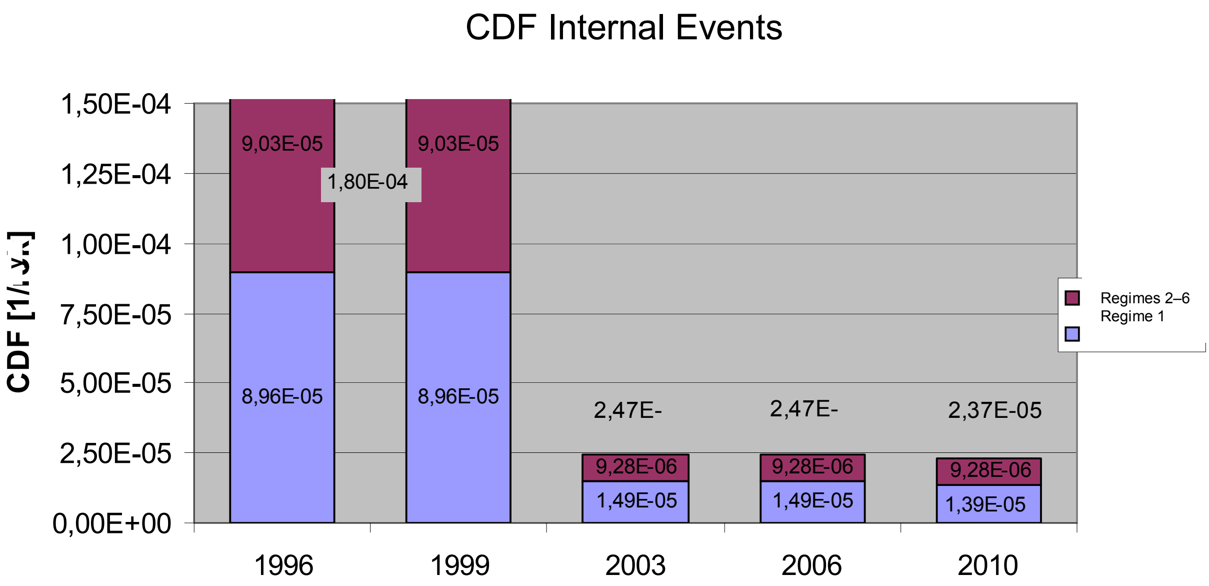


Fig. 37: History of CDF results (internal event)

CDF - average annual frequency of damage to the fuel in the core for at power and low power) operation of the NPP

The contribution from the category of large radioactive releases (LERF) to the total value of the CDF is 27.11 %.

Probabilistic models for monitoring the level of risk in real time (Safety Monitor) were elaborated in 1996 – 1999, updated in 2003 and implemented during operation. This system is used for the identification and monitoring of risk configurations for all units during shutdowns, for monitoring risk profiles in real time during operation, as well as during shutdowns of individual units. The system is also used for assessment of the risk of operation for the purpose of creating risk-informed applications.

Based on the current knowledge, while considering external design basis events, the following conclusions can be drawn from Level 1 PSA:

- Contribution of seismic events to the risk of CDF is below 1E-7/year.
- Contribution of other external initiating events to the risk is negligible (CDF of order 1.0E-7/year).
- Contribution of sequences of accidents leading to internally caused SBO, i.e. after LOSP, is of order 1E-6/year.

III.2 Earthquakes

Correct understanding of the following text requires familiarity with the contents of Chapter III.1, which describes technological systems designed to carry out main and auxiliary safety functions within the NPP Temelín.

III.2.1 Design basis

III.2.1.1 Design basis earthquake

Characteristics of the design basis earthquake (DBE)

In accordance with worldwide practice there are two design basis types of earthquakes for the NPP Temelín project:

MDE (Maximum Design Earthquake, referred to also as DBE-2 Earthquake according to the IAEA Safety Standards Series No. NS-G-3.3 and NS-G-1.6,

DE (Design Earthquake, referred to also as DBE-1 Earthquake according to the IAEA Safety Standards Series No. NS-G-3.3 and NS-G-1.6,

Tab. 27: Binding design values of acceleration

DBE	Level	Acceleration (PGA)	Duration	Comparable to $I_{state.}$
MDE	$DBE\text{-}2_{hor}$	0.1 g	4 - 8 sec.	7° MSIS-64
	$DBE\text{-}2_{ver}$	0.07 g	4 - 8 sec.	
DE	$DBE\text{-}1_{hor}$	0.05 g	4 - 8 sec.	6° MSIS-64
	$DBE\text{-}1_{ver}$	0.035 g	4 - 8 sec.	

PGA – maximum value of acceleration in horizontal and vertical direction at the level of free terrain (peak ground acceleration)

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

Regardless of the magnitude of acceleration, which follows from the assessment of the location, the project complies with recommendations of IAEA (NS-G-3.3, section 2.6) for minimum value of acceleration in horizontal direction PGA_{hor} = 0.1 g.

Methodology used to evaluate the design basis earthquake

The region of NPP Temelín is, in accordance with recommendations of IAEA NS-G-3.3, defined as the area within c. 150 km from the NPP Temelín, in south and south-east direction extended to about 230 km. This stretching of the region around the Temelín NPP is justified by the generally accepted low attenuation effect of east Alpine earthquakes as they spread into the Bohemian Massif. The region overlaps with two basic European geological units – the Moldanubian Zone and the Alpide Belt. The site of the Temelín NPP is within the Bohemian Massif, which is a part of the European Hercynian orogeny. European Hercynites are residuals of wide, wrinkled, moving zones wedged between the Eastern European, i.e. pheno-Sarmatian plate, on the east border with the epi-Caledonian plate, and the northern edge of the European Alpide belt.

The Alpide belt comprises the Alps, stretching south of the Bohemian Massif, and the Carpathian Mountains in the southeast and east. The deep contact of the Alpide belt with the structures around it is flat. It is a flat tectonic transfer of the nappes and units of the Alps and of the Carpathian Mountains on the southern border of the platform. Based on geological and geophysical evidence, confirmed by drills, the sunken Hercynian plate stretches 30 – 40 km into the Alpide belt under the Alps. The surface border is related to the reach of the wrinkled formations within the Alpine-Carpathian system, referred to as the Alpine front or Alpine-Carpathian foredeep. It stretches from Janovo via the arc of the Swiss Alps between Bern and Zurich, continues along the Danube in Austria, then towards Znojmo, Ostrava, turning under Krakow and extending south along the Carpathian Mountains, ending in the arch of the Eastern Carpathian Mountains near the Danube. The border of the Bohemian Massif with plate units is represented in the north by a number of deep plate faults.

The first assessment of the magnitude of the seismic threat to the site of the Temelín NPP was carried out in 1979. Based on the probabilistic evaluation of the catalog of earthquakes in the past, it was established that with 90% probability, the limit 5.5° MSIS-64 will not be exceeded within the design lifetime of the power plant. Based on realistic evaluation of the seismic activity within the region using various methods (seismo-static, seismo-tectonic, zone-free) and surveys carried out since the 1970s and in 1984, the following earthquake levels were specified and confirmed for the original project design:

MDE = 6° MSIS-64 with acceleration PGA_{hor} = 0.06 g

DE = 5° MSIS-64 with acceleration PGA_{hor} = 0.025 g

The design values of seismic threat for the site of the Temelín NPP were reassessed in 1995 in connection with the recommendations of the IAEA mission to evaluate the safety of the site. Based on the updated input data and recommendations from the IAEA regulations the values of MDE and DE were set.

In order to determine the design parameters of earthquakes of MDE magnitude (DBE-2), three different methods were used, with the seismo-static in two different variants. The outcome values were defined on the basis of a comparison of the results of all methodologies used as the most conservative values. Combining these methodologies should eliminate the inaccuracies in the catalog of earthquakes and the adverse generalization of schemes of epicentres, and improve the reliability of the final figures.

- Seismo-static (probabilistic) – elaborated in two methodological materials using the same catalog of earthquakes, but different sets of epicentres.
- Seismo-geological (seismo-tectonic) – based on the assumption that earthquake epicentres are connected to active faults.
- Experimental – referred to as the “zone-free method”, which does not require a definition of source zones and their delimitation, nor specification of the parameters of seismicity and their seismic potential. It is based on the measurements of real characteristics of attenuation along the line epicentre – assessed construction.

Seismo-static approach – 1st method

When assessing the seismic risk, it is assumed that the tectonic and seismic generating processes are stable; i.e. it is assumed that the trend of seismic activity observed in the past will remain stable in the future. The calculations also include the expectation that an earthquake can occur in any place of any area or active section of a fault up to a certain maximum possible earthquake magnitude for the particular area or fault.

From the safety point of view, the worst-case-scenario consequences were considered in terms of the maximum possible earthquakes in individual epicentres, as well as the nearest distance between the border of epicentres or active section of a fault and the site.

According to the original IAEA 50-SG-S1 recommendation, as well as according to the IAEA NS-G-3.3 safety standard, the following two approaches were used in determining the seismic risk of the location:

- Expert estimate based on a map of seismic regionalization.

* Probabilistic estimate based on a theoretical mathematical model.

The maximum design values of macro-seismic intensity for the site in question, in relation to the epicentre area, for the site of the Temelín NPP were found by considering the epicentre areas, magnitudes of the biggest earthquakes that could potentially be produced in these areas within a time horizon of 10,000 years, and the lines of attenuation of macro-seismic intensities constructed using the azimuths of the epicentre areas, considering the shortest distance of the epicentre area from the site (in accordance with the methodology used in nuclear engineering, this is the most conservative estimate).

The data analysis

Data analyses proved that the biggest earthquake intensities in the Temelín site for a time horizon of 10,000 years are as follows:

6.5° MSIS-64 in case of a maximum earthquake in epicentres 17 and 18

6° MSIS-64 in case of a maximum earthquake in epicentres 6 and 13

5.5° MSIS-64 in case of a maximum earthquake in epicentre 15.

<u>Seismo-static approach – 2nd method</u>

The approach using the 2nd method was based on the calculation of seismic risk using probabilistic analysis and partially also seismo-tectonic input information (probabilistic curves of seismic threat). This method allows assessing the probability of the annual occurrence of vibration movements of various magnitudes for many years to come, but also assessing the uncertainty of these values.

The prognosis of seismic events is based on the following data:

* Distribution of source zones in the location and region.
* Seismicity of source zones and the maximum possible earthquake that can occur in these zones (seismic potential).
* The decline (attenuation) of the magnitude of seismic movement, depending on the distance from the epicentre towards the site
* Definition of epicentre areas and their seismicity.

The assumption that the parameters of historical seismicity will also be valid in the future is based on the notion of repeated rough slides within existing faults. However, experience shows that new epicentres are to be found in places where no historical seismicity has been recorded. This assumption is one of the uncertainties in the input data.

Source areas of the seismic threat are epicentre areas of historical earthquakes, as well as lineaments of tectonic faults or their crossings. Within Central Europe, which roughly includes also the region of the Temelín NPP, 60 epicentre areas have been identified. The seismicity of these areas is expressed using count graphs and the values of their maximum possible earthquake (seismic potential).

Other source zones are the faults within the inner part of the Bohemian Massif, characterized by experts' estimates of seismicity parameters. In total, 71 source areas were identified from fault assessment, including seismic areas. An analysis of the seismic risk of the location of the Temelín NPP was executed on the basis of the evaluation of probabilistic occurrence of earthquakes in these source areas.

Seismo-geological (seismo-tectonic) approach

To assess the seismic activity of the faults in the area of the site, the faults were divided into three classes and six categories, with respect to the magnitude (M_{max}) of an earthquake that they have the potential to generate. The potential of the faults is assessed separately for individual structural units – regional geological units.

Tab. 28: Classification system for faults and numerical codes of the classes

Class	Verbal characterization	Category	M_{max}	I_0 [°MSIS-64]
A	Important seismo-genic line	I	6.5	9.5
		II	6.0-6.4	9
B	Important seismo-tectonic line	III	5.3-5.9	8
		IV	4.7-5.2	7
C	Seismo-tectonic line	V	4.1-4.6	6
		VI	3.6-4.0	5

The data (maximum design values of macro-seismic intensity for the site of the Temelín NPP depending on the seismo-active section of the fault) were generated by weighing the map of seismo-active faults using the values of the maximum possible intensities of earthquakes that can be produced in seismo-active sections of faults in a time horizon of 10,000 years and the attenuation curves of macro-seismic intensities, constructed for the site of the Temelín NPP considering the shortest possible distance of the seismo-active sections of faults from the site (i.e. in accordance with current methodology, it is the most conservative estimate).

The analysis of the data shows that the maximum intensity of an earthquake in the Temelín location for the time horizon of 10,000 years is 6.5° MSIS -64.

Experimental approach

Experimental determination of the seismic risk is based on the “zone-free method”. This method has a number of advantages; in particular, it does not require the definition of source zones and their borders, nor determination of seismicity parameters and the corresponding seismic potential.

This method can be used as the last method, when records of seismic acceleration in the location of the NPP are available. The new method does not have to rely solely on subjective macro-seismic data (catalogs of historical earthquakes and isoseist maps). When using authentic instrumental data the uncertainties of previous methods, stemming from various empirical conversion relationships and expert estimates, are eliminated. For example, the relationship between the acceleration of seismic oscillations and local macro-seismic intensity is burdened with uncertainties up to the second order. The new experimental method is accepted by experts as reliable and promising.

The main source of seismic threat to the NPP is the seismic effect of strong earthquakes in active seismic zones in the region, located more than 150 km from the NPP. For this purpose, it is necessary to determine the best attenuation equations for acceleration, applicable to the propagation of seismic waves within the Bohemian Massif. In the first estimate, these effects can be captured by the value of horizontal acceleration caused by the earthquake.

The assessment of seismic threat to the site of the Temelín NPP using the experimental method is an alternative method to the above deterministic and probabilistic methods. The experimental method employed works with the following input data:

– Catalog of magnitudes and coordinates of historical earthquakes

– Correction of historical magnitudes and distances of epicentres

By screening the available catalogs, those historical earthquakes were selected which, after correction, were concluded to cause $PGA_{hor} \geq 10\ cm.s^{-2}$ acceleration of soil within the site of the Temelín NPP.

III.2.1.1.1 Position of epicentres of selected earthquakes in the area of the Alps

Calculations of seismic threat to the Temelín NPP using the "zone-free method" for a time interval of 800 years have proved that the estimate of the level of seismic risk using deterministic and probabilistic methods was sufficiently conservative.

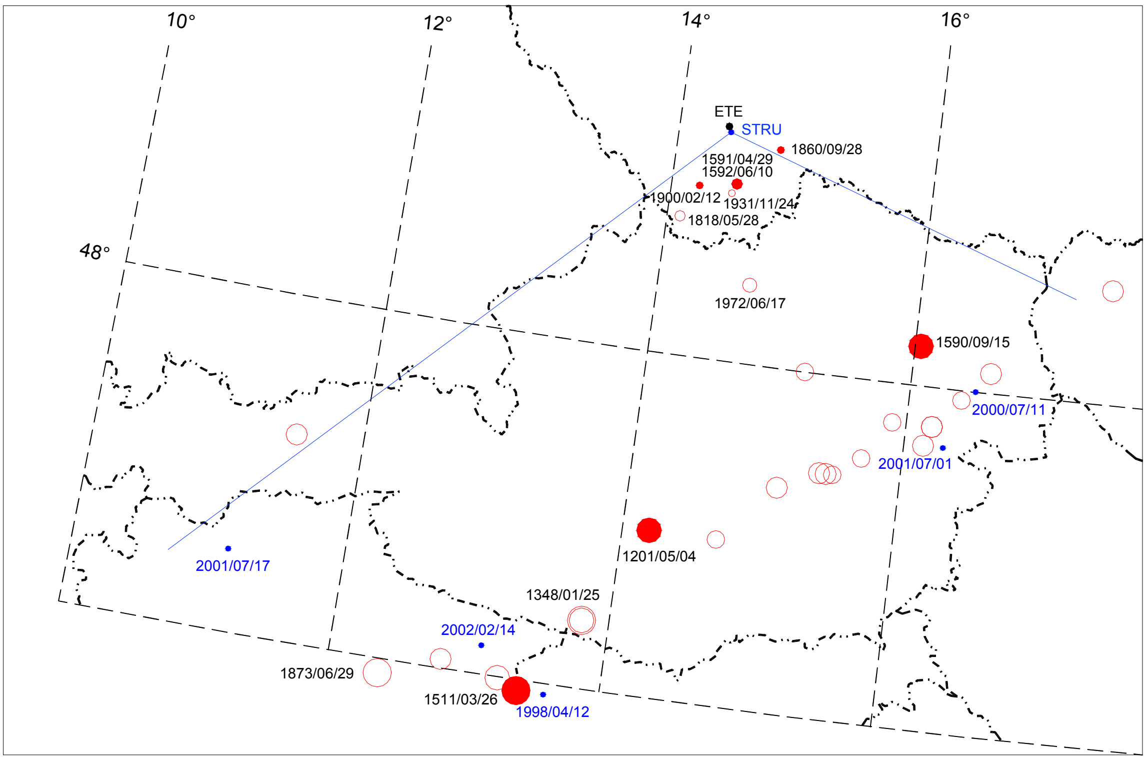


Fig. 38: Position of epicentres of selected earthquakes in the area of the Alps

III.2.1.1.2 Conclusion on the adequacy of the design basis for the earthquake

The value of the peak acceleration of the bedrock $PGA_{hor} = 0.1$ g and $PGA_{ver} = 0.07$ g for the maximum design earthquake and $PGA_{hor} = 0.05$ g and $PGA_{ver} = 0.035$g for a design earthquake ensure sufficient seismic resilience of the Temelín NPP for this site.

There are no tectonic structures within the Czech Republic that would be able to generate strong earthquakes. Evaluation of the historical data and long-term monitoring have revealed that the site of the Temelín NPP is seismically very quiet. The results from the network for detailed seismic regionalization substantiate the correctness of the conclusion of the seismic assessment of the site of the Temelín NPP. Continuous evaluation of the positions of the epicentres of local micro-earthquakes reveals that in a number of cases, the causes are related to the geological structure of the southern part of the Bohemian Massif. With 95%

likelihood, an earthquake of a magnitude exceeding 6.5° MSIS-64 (PGAhor = 0.08 g) cannot occur within the site of the Temelín NPP.

The seismic resilience of buildings and selected parts of the Temelín NPP (fragility analysis, assessment of seismic resilience, etc.) was evaluated as a part of the seismic analysis of risk. The results of the analysis of the seismic resilience of buildings and selected parts of the system proved that the real resilience of all safety systems and structures significantly exceeds the value of 7° MSIS-64 (PGA_{hor} = 0.1 g) determined for the MDE.

SSC (system, structure, component) relevant for carrying out safety functions are resilient at least up to the value of 7° MSIS-64 (PGAhor = 0.1 g), which provides a sufficient safety margin for the 5% uncertainty. There are differences in the resilience of individual SSC; however, these contribute to the increase of the safety margin for fulfilling safety functions.

The units of the Temelín NPP are fitted with seismic monitoring system. Individual earthquake magnitudes are assigned corresponding action levels for announcing incidents and the activation of OER (organization of emergency response). The staff in the NPP is sufficiently qualified and trained to assess the damage to the system after a seismic event.

The data related to the assessment of the seismic risk in the location are updated regularly on the basis of measurements from stations of local seismologic network for detailed seismic regionalization.

The outcomes of monitoring up to now can be summarized in the following points:

- Within 40 km around the Temelín NPP, no earthquake with a magnitude exceeding 1 has occurred.
- Within 50 km around the Temelín NPP, only 9 micro-earthquakes with magnitudes of 1-2 have occurred, and no earthquakes with higher magnitudes.
- By evaluating industrial explosions in quarries within the site,it was verified that the network is capable of detecting and localizing tremors with magnitudes of 1 – 3 within 50 km around the Temelín NPP.

III.2.1.2 Provisions to protect the plant against a design basis earthquake

III.2.1.2.1 Identification of the SSC that are required for achieving a safe shutdown state

Building constructions and technological systems that are necessary for carrying out basic safety functions (controlling reactivity, transferring heat from the reactor core, capturing ionization radiation and radionuclides) during earthquakes, as well as constructions and systems whose disruption or failure during an earthquake could endanger other constructions and systems nearby which are important for nuclear safety, are classified in the 1st category of seismic resilience.

In order to provide a more accurate assessment of the impact of seismicity on the systems and facilities within Category 1 of seismic resilience, these are divided into the following subcategories:

(1) Subcategory 1a - Full functionality must be maintained up to the level of the MDE (inclusive).

(2) Subcategory 1b - Only mechanical stability and air-tightness are required up to the level of the MDE (inclusive).

(3) Subcategory 1c - Seismic resilience only from the point of view of possible seismic interactions is required; in particular, a stable position up to the MDE (inclusive) must be maintained. The aim is to prevent impact on the systems within Subcategories 1a and 1b.

III.2.1.2.2 Evaluation of the SSC sturdiness in connection with the DBE and assessment of the potential safety margin

Building constructions and technological systems in the 1st category of seismic resilience were seismically analyzed. These analyses included safety relevant buildings, components, service systems, I&C (oversight and control system) and electrical devices and were conducted either by experiment, calculation or indirect assessment. The outcome reveals their resilience corresponding to the MDE stress values.

Tab. 29: Resilience parameters ("Fragility") of the selected type of system

Type of system	HCLPF
Insulators of external 400 and 110 kV switchyards	0.10 g
Supply fuel storage for DGS	0.11 g
400 and 110 kV switchyards	0.13 g
0.4 kV switch boxes for safety systems	0.15 g
6kV switchboards for safety systems	0.15 g
6kV/400V transformers for safety systems	0.15 g
Cooling tanks with injection	0.17 g
0.4 kV boxes of DG distributors	0.18 g
Ventilators	0.18 g
Tanks with pure boric concentrate	0.19 g
Emergency supply tanks for SG	0.19 g
ECCS exchanger	0.23 g
Tank for high pressure ECCS	0.28 g
Tank for the system of emergency injection into the I.C	0.46 g
Tanks for the spray system	0.46 g

Calculation of the HCLPF (High Confidence on Low Probability Failure) is based on the concept of bedrock acceleration leading with "high confidence to low probability failure". The HCLPF value of bedrock acceleration is, therefore, acceleration of a particular construction object or system that gives 95% probability that the likelihood of failure will be less than 5%.

The boxes of the I&C safety systems are qualified for acceleration of at least 0.3 g.

The outcomes of the "fragility analysis" of buildings and selected systems within the Temelín NPP prove that the resilience of all safety-relevant systems and construction objects significantly exceeds the value of 0.1 g determined as the MDE.

III.2.1.2.2.1 Main operating provisions to achieve a safe shutdown state

Measurements of local detailed seismic regionalization (DSR) of the Temelín NPP are also used to support the outcomes of the evaluation of seismic threat in the site of the Temelín NPP.

The main purpose of the network for detailed seismic regionalization of the Temelín NPP is to register local micro-tremors with magnitudes within the range of 1 - 3. Seismic events are registered in 4 categories: Teleseismic events more than 2,000 km away, regional events in the range of 200 – 2,000 km, close events in the range of 50 – 200 km, and local events closer than 50 km. In addition to tectonic earthquakes, the network of stations also registers induced mine tremors and industrial explosions.

The network of detailed seismic regionalization near the Temelín NPP, built and operated according to the recommendations of the IAEA, has been in continuous operation since September 1, 1991. Seismic events are registered in 4 categories: Teleseismic events more than 2,000 km away, regional events in the range of 200 – 2,000 km, close events in the range of 50 – 200 km, and local events closer than 50 km. In addition to tectonic earthquakes, the network of stations also registers induced mine tremors and industrial explosions.

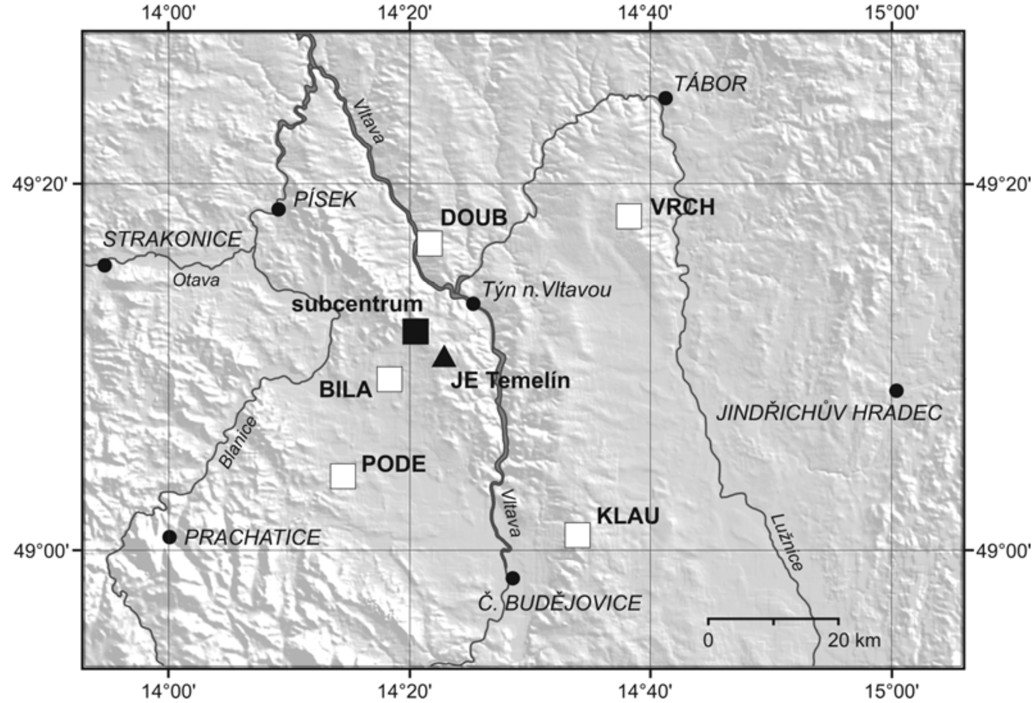


Fig. 39: Map of stations within the DSR network at the Temelín NPP

The units of the Temelín NPP are equipped with a seismic monitoring system (SMS). The SMS is activated any time the pre-set limit values of acceleration (0.005 g in the horizontal and vertical directions for sensors in the ground and in the foundation, 0.015 g in the horizontal and 0.045 in the vertical direction for the sensor in the containment) are exceeded. Together with the SMS, the corresponding alarm systems within the safety systems are also activated. No initiation signals are sent into control or safety systems in the NPP units when the SMS or earthquake alarms are activated. After each seismic event, the condition of the units must be thoroughly assessed. Controlled shutdown of a unit is required if the design earthquake level was exceeded or if signs of seismic damage were detected even if the design earthquake level was not exceeded.

The individual earthquake levels are also assigned the corresponding response levels for announcing extraordinary events and activation of the OER. The staff of the NPP is sufficiently qualified and trained for EOP and SAMG, as well as for assessing the damage to the systems after a seismic event.

It has been confirmed, for all earthquakes that need to be considered in the site of the Temelín NPP, that basic safety functions will be fulfilled:

a) Controlling reactivity

b) Transferring heat from nuclear fuel

c) Capturing ionizing radiation and radionuclides.

Failure to shut down the reactor due to mechanical failure of the falling cluster system has the nature of an ATWS scenario, which was analyzed for the Temelín NPP and for initiation events that can appear as a consequence of an earthquake. The reactor should be SHUT DOWN by feedback effects of reactivity – the injection of boron by at least one of the three systems for emergency boron injection.

Transfer of residual heat from the reactor in a hot and semi-hot state would be carried out in the secondary "feed and bleed" mode by adding water to the SG via the emergency supply system for the SG and by controlled release of steam into the atmosphere up to the temperature of 150 ^{o}C within the I.C. To reach this temperature, there is a sufficient amount of water stored in the tanks. Subsequently, the heat transfer would continue via one of the three redundant lines for the transfer of residual heat into the atmosphere – via SIS exchangers cooled by the ESW and via the CBSS.

If the above means of heat transfer from the reactor core cannot be used, there is the possibility of alternative heat transfer from the reactor core using the method of primary "feed and bleed"; controlled drainage of coolant from the I.C into the containment and then via ECCSexchangers cooled by the ESW and via the CBSS into the atmosphere.

Heat from the SFSP would be transferred using one of the three routes of the SFSP cooling system to the corresponding exchangers cooled by the ESW and through the CBSS into the atmosphere.

III.2.1.2.2.2 Protection against indirect effects of an earthquake

The indirect effects of earthquakes were assessed for the site of the Temelín NPP. None of the assessed effects poses a significant threat to the NPP.

1) Internal floods in the NPP were evaluated as a part of the analyses of probabilistic assessment of the risk of flood. Those flood scenarios, in which floods were to contribute significantly to the risk of damage to the reactor core and subsequently to the risk of large releases of radioactive substances, were analyzed and their contribution to the overall risk of damage to the reactor core due to internal floods was quantified. The validity of the assumptions in these analyses was verified by physical inspection, which revealed no facts contradicting the original assumptions used in the analyses of the risk of internal floods or would lead to a change in these assumptions.
2) A seismic event could damage seismically non-resilient (fragile) systems and objects, which could lead to disconnection of the NPP from the electrical grid and the supply of working media. The project was designed to cope with a loss of the external power supply (400 kV and 110 kV) and with failure to regulate the TG for house consumption (full loss of the power supply for house consumption). As a consequence of this situation, both reactors would be shutdown and the heat from the reactor cores would be transferred by natural circulation of the primary coolant. The power supply for the above safety functions would be provided from emergency power sources (DG + accumulators), which are seismically resilient and located in seismically resilient buildings. The operating inventory of diesel fuel in seismically resilient objects is sufficient for several days of operation of the DG. More diesel fuel would be supplied by tank trucks.
3) In case of a seismic event, the Hněvkovice raw water pumping station could also be lost. This station, and the pipe systems bringing water into the NPP, are not seismically resilient. The systems for heat transfer are seismically resilient and located in seismically resilient objects. The water supply in the location (CBSS) is sufficient for transferring heat from the reactors and pools of spent fuel for at least 3 x 12.5 days.
4) In the event of major destruction within the infrastructure and the long-term unavailability of the site (collapse of buildings, damaged communications etc.), if the new staff could not reach the location in the initial days, the necessary activities (inspection of the facilities and shutdown of the units) would be carried out by the staff present at the

location at the time of the event. Replacement would be carried out in cooperation with the authorities of the state administration (integrated rescue system, army, etc.).

5) Access to the key buildings could be limited as a consequence of the destruction of seismically non-resilient objects along the routes of internal access communications as well as due to debris falling into the access area of the power plant. Furthermore, access to the emergency control centre and shelter for emergencies could be blocked. In this case, it is possible to use backup access/entry into the compound and activate the emergency control centre in České Budějovice. If the shelters are not accessible, non-essential personnel would be evacuated from the site; essential personnel would carry out activities from locations within seismically resilient objects. The technical support centre would be working from the unit or emergency control room.

6) As an early indication of internal floods, an indirect consequence of earthquake, water detected in rooms is signaled to the CR (unit control room). Actions to be taken by the staff of the NPP if water is indicated in certain rooms are defined in the corresponding procedures.

III.2.1.3 Compliance of the plant with its current licensing basis

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

The current state of the facilities of the NPP is in accordance with the design requirements. In order to keep the current state of the facilities of the NPP in accordance with the designrequirements, a number of regular activities are carried out, including:

- Maintaining the seismic qualification of the systems and buildings
- Walkdowns to ensure the required condition of the facilities, to prevent damage, accidents and fire or injuries to persons and to ensure a high safety of operation.
- Operating checks and tests of systems.
- Predictive and corrective maintenance of the systems.

Licensee's processes to ensure that mobile equipment and supplies are in continuous readiness for use

A fire rescue unit of approximately 16 firemen is on standby in the site of the Temelín NPP (permanent, 24 hours). The unit has the necessary equipment and is trained to respond and bring under control any fire in any place within the compound. Fire equipment and response personnel are stationed in a building that is not seismically qualified. To reduce the risk of injuries and damage, all equipment and response personnel will be moved to free areas in case tremors and other indirect symptoms are registered.

The fire rescue unit has 4 tank trucks with fire pumps, 1 combined fire truck and 3 trailer fire pumps with a total nominal output of 280 l/s.

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

No significant deviations of the current state from the design requirements were found during special inspections aimed to verify seismic resiliency carried out after the events in the Fukushima NPP in May 2011.

III.2.2 Evaluation of safety margins

III.2.2.1 Range of earthquakes leading to severe fuel damage

In practice, an earthquake in the site of the Temelín NPP with a magnitude exceeding 6.5° MSIS-64 (PGA_{hor} = 0.08 g) is impossible. SSC fulfilling safety functions are resistant up to the value of 7° MSIS-64 (PGA_{hor} = 0.1 g).

Earthquakes with a magnitude in excess of the MDE (i.e. higher than 7° MSIS-64), are not "cliff-edge" limiting conditions due to the individual safety reserves and internal redundancies of the VVER1000 project. "Fragility analyses" carried out for objects and selected systems in the Temelín NPP prove that the resilience of the safety relevant systems and construction objects significantly exceeds 7° MSIS-64.

There are specific differences in the resilience of individual SSC; however, these further increase the safety margin, ensuring that safety functions are fulfilled. Fulfillment of the safety functions ("cliff edge") could be threatened if the acceleration values exceeded PGA_{hor} = 0.15 g, which is not realistic in the site of the Temelín NPP. By approximating the lines of seismic threats for the location into higher intensities, the frequency of the occurrence of seismic events with an intensity of PGA_{hor} = 0.15g and higher can be estimated to be less than 1E-8/year. This corresponds to an occurrence of once in 100 million years or less.

The integrity of the SFSP in the event of an earthquake would not be threatened (these tanks are located in a seismically resilient containment, surrounded by reinforced concrete with stainless steel lining; the bottoms of the pools are at +20.7, the upper rims at +36.9 m). In case the SFSP cooling function fails or the coolant is leaking, it is possible to resupply the pools and cooling system using a dedicated supply line from the outlet of the pumps of the containment spray system, by draining coolant into the containment and by the exchangers of the SIS (safety injection system) cooled by the EWS and via the CBSS into the atmosphere.

Range of an earthquake leading to a loss of containment integrity

The barriers preventing radioactivity leaks – covers on the fuel, pressure boundary and the protective containment - are seismically resilient. The resilience of the concrete construction of the containment (prestressed and reinforced concrete) certainly exceeds the value of the design seismic resilience. The isolation of the pipelines and valves is ensured by redundant separating components with resilience of at least 0.1 g (with sufficient reserve). The resilience of the I&C (oversight and control system) for automatic isolation of the containment corresponds to the value of 0.3 g. Based on an engineer's estimate of the project architect, Level 0.2 and higher can be assumed. In any case, the resilience considerably exceeds any possible earthquakes that could occur within the location.

III.2.2.2 Earthquake exceeding DBE and consequent flooding

The location is certainly not threatened by floods as a consequence of an earthquake. The main buildings in the Temelín NPP that contain systems relevant for nuclear safety are 507.30 m above the sea level. This is 135 m above the level of the Hněvkovice water reservoir, which is a part of the Vltava cascade with regulated flow.

The Temelín NPP was subject to a safety assessment also with respect to the potential destruction of water reservoirs in the upper parts of the Vltava River (including an earthquake). There are only two large reservoirs above the Hněvkovice reservoir; Lipno I on the Vltava and Římov on the Malše, and one smaller reservoir, Lipno II.

If the largest reservoir, Lipno I, breaks down, the flood wave would culminate in the Hněvkovice reservoir at 376.7 m above sea level. That corresponds to a 1 in 10,000 year flow in the profile of Hněvkovice. Although this would lead to a loss of the Hněvkovice raw water pumping station, it has been shown that the water inventory in the site is sufficient for providing heat transfer from the reactors and spent fuel pools for at least 3 x 12.5 days.

III.2.2.3 Measures that can be foreseen to increase the sturdiness of the plant against earthquakes

As clearly shown in the assessment, the site of the Temelín NPP was chosen exceptionally well from seismic point of view. The site can be characterized as highly stable in relation to external natural events, including seismicity. Moreover, the sturdiness of the VVER1000 project and diversity of the seismically resilient SSC ensure a sufficient resilience and safety margin in case of design and beyond design seismic events.

The potential adverse effects of earthquakes are, therefore, limited only to seismically non-resilient SSC, which may be contributing to the fulfillment of auxiliary safety functions. This is the case of, for example, a long-term power supply after losing the external power supply (3 days and more) using just emergency sources, which require external resupplying with diesel fuel for the DG.

Activities after a seismic event could also be complicated by a loss of the means of communication between the control centres and responding persons including the communication with external control centres and state administration due to damaged infrastructure around the NPP.

The aim of the proposed measures is further strengthening of defence-in-depth protection in case of earthquakes. The opportunities to improve the defence-in-depth protection are listed in the following table. The table also contains areas that require the execution of additional analyses, because they were not available at the time this assessment was executed.

Tab. 30: The opportunities to improve the defence-in-depth protection against earthquakes

Opportunity for improvement	**Corrective measure**	**Time period** (Short-term I / Medium-term II)	**Note**
Technical means	Alternative refueling diesel using tank trucks for long-term operation of the DG	I	
Regulations	EDMG manuals for using alternative means	II	
Emergency readiness	OER (organization of emergency response) ability outside the ECC (emergency control centre)	I	
Analyses	Resilience of the LFRU (local fire rescue unit) to seismicity	I	
Communication	Alternative means of communication after a seismic event	I	
Staff	Analysis of the threat to the shelters in case of a seismic event	II	
Staff	Security of the staff after a seismic event	I	
Analysis, machinery	Access to buildings, accessibility for heavy machinery	II	

III.3 Flooding

Correct understanding of the following text requires familiarity with the contents of Chapter III.1, which describes the technological systems designed to fulfill main and auxiliary safety functions within the Temelín NPP.

III.3.1 Design basis

III.3.1.1 Flooding against which the plant is designed

Characteristics of the design basis flood (DBF)

The Temelín NPP is located in an area with normal precipitation, with no history of extremely high precipitation. From a drainage point of view, the site of the Temelín NPP takes the form of cascades, with objects relevant for nuclear safety located on the highest ground, 507.10 m above sea level, with a slope decreasing towards the edge of the location, which allows natural gravitational draining from the compound even when sewerage system fails.

Design values for flooding of the location in case of maximum precipitation with an occurrence rate of once in 100 and 10,000 years are based on the assumption that water is drained from the area of the power plant on the surface, with the sewerage system completely disabled due to blocked inlets. For the assessment of the real hydrologic characteristics of an area with the final highest level of 507.10 m above sea level from a flood safety point of view, the decisive factor is the maximum aggregate daily precipitation. The area in question would, in this case, be exposed to 47.2 mm at once-in-100-years precipitation and 88.1 mm at once-in-10,000-years precipitation. An area with a final terrain modification 504.10 m above sea level will drain extreme precipitation over the edge with a maximum height of 114 mm.

The main buildings within the Temelín NPP containing systems relevant for nuclear safety are 135 m above the level in the Hněvkovice reservoir, which is a part of the Vltava cascade with regulated flow. The Hněvkovice water reservoir was constructed as the main component in a system supplying the Temelín NPP with technological water. In practical terms, the site cannot be flooded as a consequence of increased flow in the profile of the Vltava River.

Methodology used to evaluate the design basis flood

From the assessment of floods near watercourses, it follows that in the profile of the Hněvkovice 10,000-year flood, water will raise to a level of approximately 5 m above the maximum level, which will result in flooding of most of the raw water pumping station for the Temelín NPP. Subsequently, the Hněvkovice water reservoir may collapse as well. Both of these conditions will disable standard operation and raw water supply for the Temelín NPP; therefore, both units in the Temelín NPP will have to be shut down. However, the amount of water stored in the location is sufficient for cooling the units to a cold state. Water reserves are stored in the water reservoir, in the tower cooling system and, last but not least, there is also the possibility of supplying the service water system from the potable water system.

The assessment of large flow rates within the profile of the Hněvkovice water reservoir was verified by the floods in 2002. The maximum level reached on August 13, 2002, was at the level of 371.56 m above sea level, i.e. reaching a height that is near the considered long-term maximum for this water reservoir (371.60 m above sea level).

The assessment of large flow rates within the profile of the Hněvkovice water reservoir was verified by the floods in 2002. The maximum level reached on August 13, 2002, was at the level of 371.56 m above sea level, i.e. reaching a height that is near the considered long-term maximum for this water reservoir (371.60 m above sea level). Water passed through the Hněvkovice dam in a standard way, no significant damage was found on the reservoir or the pumping station for the Temelín NPP.

The design of the rain sewerage system (conducting precipitation water) in the Temelín NPP was based on a model of 15 minutes of unreduced rain with intensity of 127 $l.s^{-1}.ha^{-1}$. The site of the Temelín NPP has a drainage area of 133.14 ha, split between two collectors (Collector “A” collects water from the west and south sections with a combined area of 80.06 ha; Collector “B” collects water from the north and east sections with a combined area of 53.08 ha). The average drainage coefficient is 0.415 and the total drainage amount at nominal intensity of unregulated rain is 7 020 $l.s^{-1}$ for 15 min.

The Hněvkovice concrete gravity dam creates a water reservoir with a total volume of 22.2 $mil.m^3$ with a maximum level of 370.50 m above sea level. All critical buildings are designed for the maximum level at 372.0 m above sea level. With a standard long-term level of 365.0 m above sea level the volume of water in the reservoir is 12.8 $mil.m^3$. With the bottom of the reservoir near the body of the dam at 354.0 m above sea level, the depth of water in the location of the water intake for the Temelín NPP is between 11 and 16.5 m. The main purpose of the water reservoir is to create a damper space in case of increased flow rates from the Lipno reservoir, located 120 km upstream.

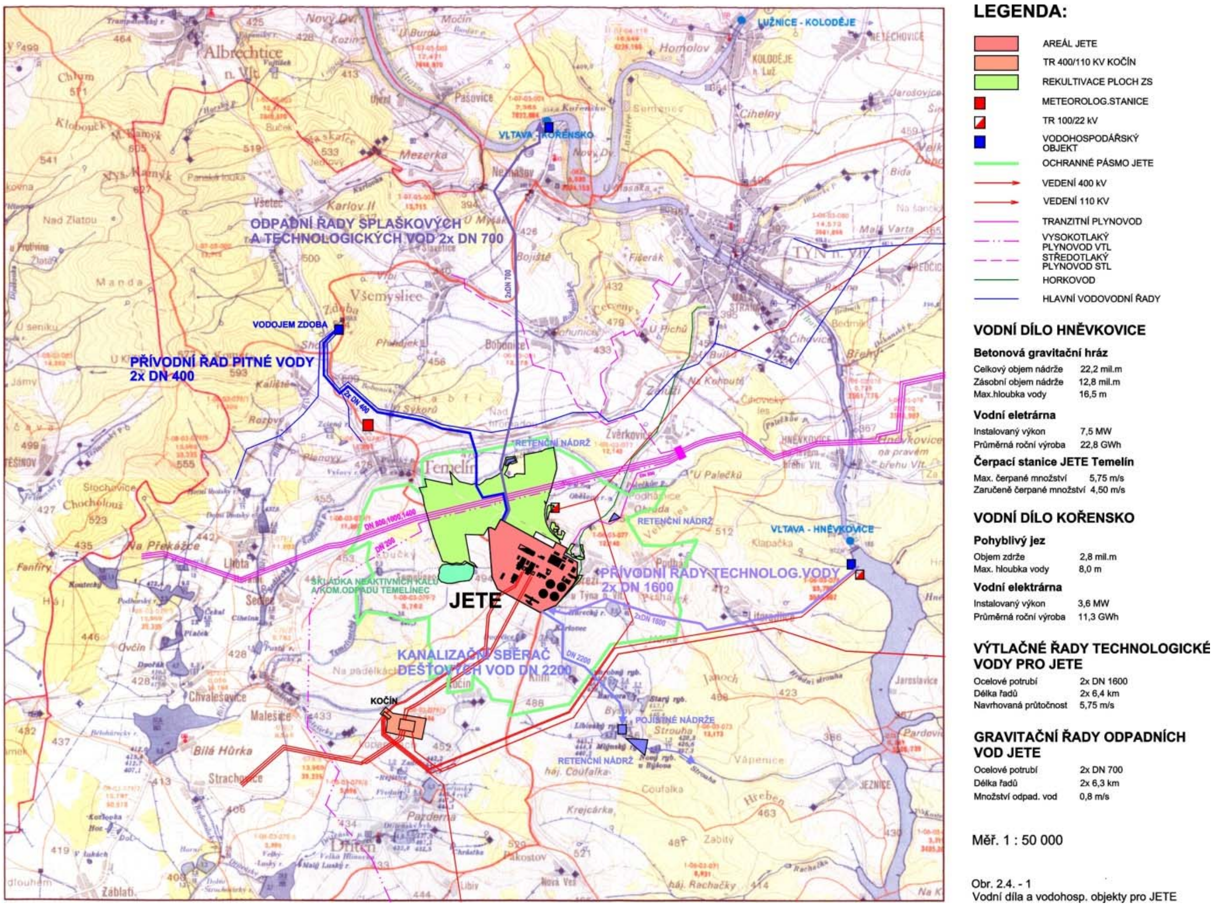


Fig. 40: Water reservoir

A decrease of the water level in the Hněvkovice reservoir below a certain level is an impulse for increasing the flow from the Lipno water reservoir so that the required water supply for the NPP is ensured in all states of operation. The reserve volume at Lipno is 252.0 mil m^3.

Above the Hněvkovice water reservoir, there are two large reservoirs; Lipno I on the Vltava River and Římov on the Malša, and a smaller water reservoir, Lipno II. In case the Lipno I reservoir is damaged, the resulting flow in the profile of the Hněvkovice would be around the level of a 10,000-year flood, which could lead to damage to the Hněvkovice reservoir. During culmination of the flood wave, the water level in the Hněvkovice reservoir would reach 376.7 m above sea level.

The Hněvkovice reservoir is very important in the winter mode, when thanks to its depth, it creates conditions for a safe water supply for the Temelín NPP under any operating and climatic conditions. An additional function of the Hněvkovice reservoir is the production of energy in the hydro power plant, which is under half-peak operation with daily balancing of the natural flow rates after deducting the water supplied by the pumping station to the Temelín NPP. The HPS is considered to be one of alternative auxiliary sources to supply the house consumption of the Temelín NPP in case of an SBO (station blackout).

The main purpose of the Kořensko weir is to maintain the water level in the end part of the Orlík water reservoir at 353.0 m above sea level, i.e. near the maximum level in the Orlík reservoir, regardless of fluctuation in this reservoir. With a level near the normal, i.e. 353.0 m above sea level, the weir reservoir contains 2.8 mil.m^3.

Similarly to the Hněvkovice water reservoir, the Kořensko water reservoir is used as an small HPS working in tandem with the HPS Hněvkovice. One of the main purposes of this construction is to create conditions for safe homogenization (dilution) of released waste water from the Temelín NPP. After flowing through the small HPS, waste water is led via an absorbing object of waste processing into the profile of the Kořensko reservoir and ends at the suction intake of the turbines of this hydro power station. In case the power plant is out of operation, waste water is released into the outlet of the weir section.

Near the Temelín NPP, the small streams Rachačka, Strouha, Hradní strouha, Palečkův potok, Bohunický potok, Karlovka and Temelínecký potok are found, which surface in the area of the Temelín NPP or very near it. Long-term monitoring has revealed no states that could lead to flooding in the area of the Temelín NPP. This claim is supported by the morphology of the area, by a radical decline of these streams in the direction away from the power plant and by the hydrological data.

The underground water levels in the NPP Temelín site oscillate between 10-12 m under the terrain, i.e. around the level of 500.0 m above sea level. Because the Temelín NPP is located on a plateau and underground water is fed only by precipitation, underground water drains away from the site of the Temelín NPP in all directions. There is no risk for the buildings or rooms containing safety relevant systems from the shallow horizon of underground water.

Conclusion on the adequacy of protection against external flooding

Considering the previous assessment and history of floods in the region, protection measures protecting structures against floods from watercourses in the site of the Temelín NPP are unnecessary. The site of the Temelín NPP is not threatened by floods from watercourses.

For the site of the Temelín NPP, it is necessary to consider only external floods caused by extreme precipitation. All SSC (system, structure, component) located in above-ground buildings at 507.10 m above sea level or higher are not under threat from external flooding (with the exception of possible flooding of the diesel storage system for the diesel generator station) due to design measures (thresholds, sealing of the access and installation openings), which ensure that accumulated water cannot flow into the buildings. All safety relevant buildings are protected against floods even in case of a maximum level of 88.1 m, which corresponds to 10,000-year precipitation per day. In case theoretically possible shorter precipitation periods with higher intensity occur, the system’s passive gravity sewerage is capable of draining the water, thanks to the large size of the sewers and short duration of these intense rainfalls.

III.3.1.2 Provisions to protect the plant against a design basis flood

Identification of the SSC that are required for achieving a safe shutdown state

The SSC necessary for achieving safe shutdown of units are described in chapter III.1. Due to measures, which ensure that accumulated water cannot flood objects containing these SSC, none of these systems is endangered in case of a flood possible for the site of the Temelín NPP. Thus, the fulfillment of basic safety functions is confirmed.

Accumulated water could endanger only those SSC contributing to the fulfillment of safety functions that are located below the level of finished terrain modifications – under a level of 507.10 m above sea level (pumping station of the ESW, resupplying diesel into the operating tanks of the DG). In case the diesel fuel management system for the diesel generation station, which supplies diesel from the reserve tanks into the operating tanks, is flooded, the long-term operation of the DG could be threatened.

The water reserve for heat transfer (auxiliary safety function) is safe in case of external flooding. Extreme floods in the profile of the Vltava River may lead to loss of the Hněvkovice pumping station, which supplies raw water for the compensation of vaporization during heat transfer into the atmosphere. Even if the raw water supply for the NPP is interrupted, the location contains a sufficient reserve of water to ensure the transfer of heat into the ultimate heat sink from the fuel in the RC, as well as from the spent fuel stored in the SFSP for at least 3 x 12.5 days (using only safety systems).

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

With respect to the possible threat from floods caused by extreme precipitation, all design measures lead to ensuring sufficient resiliency of the structures against possible floods caused by extreme precipitation.

Individual buildings have specified requirements on resiliency against accumulated water, which ensure that the access and installation openings prevent water from entering the structures or additional measures have been implemented, preventing accumulated water from accessing the buildings (thresholds).

In case water floods the pumping stations of the EWS, these contain a system for removing sludge water from the reservoir in the lowest level in the pumping station (-7.10 m).

Main operating provisions to prevent flood impact to the plant

For all flood situations possible for the site of the Temelín NPP, the fulfillment of basic safety functions has been confirmed:

a) Controlling reactivity

b) Transferring heat from nuclear fuel

c) Capturing ionizing radiation and radionuclides.

The reactor is also shutdown in case of floods by the clusters falling. This function cannot be threatened by floods.

In addition, heat transfer cannot be threatened by floods. The transfer of residual heat from the reactor in a hot or semi-hot state would be carried out in the mode of secondary “feed and bleed” by adding water into the SG via a system for normal or emergency water supply for the SG (or in other ways, according to the EOP) and the regulated release of steam into the atmosphere up to the temperature of 150 ^{o}C in the I.C. Subsequently, heat transfer would continue via one of the three lines of the system for residual heat transfer into the atmosphere, through SIS exchangers cooled by the EWS and via the CBSS.

Heat would be transferred from the SFSP via one of the three routes of the SFSP cooling system, through exchangers cooled by the EWS and via the CBSS into the atmosphere.

The barriers against radioactivity release –fuel cladding, pressure boundary of the primary circuit and containment – cannot be threatened by floods. The containment is a passive system, which cannot be threatened by floods, and the isolation of its piping and passageways is ensured by redundant separating components, which also cannot fail as a consequence of floods.

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

If the location is inaccessible as a consequence of regional floods, the activities necessary for deactivating the reactors and keeping them in a safe state would be carried out by staff present in the site at the time of the event. Replacing the staff would be ensured in cooperation with the bodies of the state administration (integrated rescue system, army, etc.).

III.3.1.3 Plant compliance with its current licensing basis

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

The current condition of the systems is in accordance with design requirements. In order to maintain this compliance, a number of regular activities are carried out, including:

- Walkdowns to ensure the required condition of the facilities and prevent damage, accidents and fire or injuries to persons and to ensure high safety of operation.
- Operating checks and tests of systems.
- Predictive and corrective maintenance of the systems.

The rain sewerage system is inspected and maintained on a regular basis. The technical condition of the sewerage routes is inspected once per year and the necessary repairs are carried out depending on the condition. The end shaft near the border of the compound is inspected once per week to check that the grating is clean.

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

A fire rescue unit of around 16 firemen is on standby in the site of the Temelín NPP for each shift (permanent, 24 hours). The unit has the necessary equipment and is trained to respond and bring under control any fire in any place within the compound. Fire equipment and response personnel are stationed in a building that cannot be damaged during floods.

The mobile equipment of the fire rescue unit (independent means for transporting and pumping media) is also designed for removing water during floods. The fire rescue unit has 4 tank trucks with fire pumps, 1 combined fire truck and 3 trailer fire pumps with a total nominal output of 280 l/s.

Potential deviations from the licensing basis and actions to address these deviations.

No significant deviations of the current state from the design requirements were found during special inspections aimed at verifying the resilience against floods carried out after the events in the Fukushima NPP in May 2011.

III.3.2 Evaluation of the safety margins

III.3.2.1 Estimation of the safety margin against flooding

The site of the Temelín NPP has never been, and is not now, threatened by floods from watercourses. The main objects of the Temelín NPP containing systems relevant for nuclear safety are located 507.30 m above sea level. This is 135 m above the level of the Hněvkovice water reservoir on the Vltava River. A safety evaluation with respect to the potential breaking of dams on water reservoirs in the upper part of the Vltava River (Lipno I on the Vltava, Římov on the Malša) was carried out for the Temelín NPP. In case the Lipno I reservoir is damaged, water approximately equaling a 10,000-year flood will flow through the profile of Hněvkovice.

In case of a 10,000-year flood, the level reached in the profile of Hněvkovice will lead to flooding of most of the pumping station supplying raw water for the Temelín NPP, which will disable the standard raw water supply for the Temelín NPP, and both units will have to be shutdown. However, the site contains a sufficient reserve of water to cool down the units to a cold state. During the biggest floods on the Vltava River in 2002, the profile of Hněvkovice recorded a level corresponding to the maximum level considered for this water reservoir. Water was passing through the dam in a standard way and no significant damage was found on the pumping station for the Temelín NPP or on the dam.

Buildings relevant for safety cannot be flooded from the gravity sewer system even in case of extreme precipitation. The Temelín NPP is built as a cascade, with buildings relevant for nuclear safety located in the highest areas and the terrain sloping towards the edges of the location, which also allows for natural gravity drainage if the rain sewer fails. The building objects in the Temelín NPP are also designed to be flood-resilient in case of a maximum one-day rainfall that leads to a maximum water level of 47.2 mm (in case of 100-year rainfall) and 88.1 mm (in case of 10,000-year rainfall), in case the sewer system is completely disabled. The location also contains mobile equipment of the fire rescue unit, which is adapted for pumping water from local floods in excess of 10,000-year values.

Because flooding from external watercourses is inherently ruled out and the building objects in the Temelín NPP are designed to be resilient against floods even in the case of extreme rainfall (watertight lids, height of entry and installation openings), there is at least 100% reserve before reaching levels at which water would flood the buildings. Thanks to the gravity drainage of water from the location, this level cannot be reached.

III.3.2.2 Measures that can be foreseen to increase the sturdiness of the plant against flooding

This assessment clearly indicates that the site of the Temelín NPP is exceptionally well-selected from the point of view of flooding safety. External flooding on the Vltava River could lead to the loss of the pumping station in Hněvkovice and thus to the loss of ability to supply raw water; however, this does not pose a threat to the safety of the NPP. The design of the project and diversity of the SSC provide sufficient resiliency and reserves in case of design and beyond design floods.

Possible negative effects of floods are limited to the fulfillment of auxiliary safety functions. It was concluded that there is a risk that the pumps for resupplying diesel will be flooded in case of long-term extreme rain. This could have an impact on the long-term operation of the emergency power sources (DG), provided that all other internal and external sources of alternating current are lost at the same time.

The aim of the proposed measures is to further strengthen defence-in-depth in case of floods. The opportunities to improve defence-in-depth are listed in the following table. The table contains also areas that require additional analyses, which were not available at the time of assessment.

Some of the measures (with a note: “findings of the PSA”) would be implemented even without this assessment, which confirmed the effectiveness and correctness of previous decisions to implement measures leading to increased resilience of the original design

Tab. 31: The opportunities to improve defence-in-depth against flooding

Opportunity for improvement	**Corrective measure**	**Time period** (Short-term I / Medium-term II)	**Note**
Technical means	Increasing the resilience of the DG in case of external flooding	I	Finding of PSR
Emergency readiness	Ability of the OER to function via the ECC	I	
Regulations	EDMG manuals for using alternative means	II	
Analyses	Analysis of the threat to the shelters in case of floods	II	

III.4 Extreme weather conditions

Correct understanding of the following text requires familiarity with the contents of Chapter III.1, which describes the technological systems designed to fulfill the main and auxiliary safety functions within the Temelín NPP.

III.4.1 Design basis

Reassessment of weather conditions used as the design basis

The load caused by natural events is based on statistical processing of the data time series of at least a 30-year period of measurements of these events in the area of the Temelín NPP, or in an area with a similar landscape. The design load of the climatic conditions assumes a frequency of occurrence once in 100 years. The extreme case of maximum design weather load considers a frequency of occurrence once in 10,000 years. Buildings of the 1st seismic category must withstand the effects of extreme design conditions without posing risk to the functioning of systems relevant for nuclear safety. Other buildings must withstand the design level of weather conditions.

The following extreme weather effects, relevant with respect to site of the NPP, were considered in the assessment of resiliency of construction objects and systems against other natural events:

(1) Wind

(2) Snow/ice

(3) High/low temperature.

High winds

Assessment of the load is based on the measured annual maximum values of wind speed. The values determined on the basis of measurements carried out at the Prague-Ruzyně station, i.e. 49m/s for a frequency of once in 100 years and 68 m/s for a frequency of once in 10,000 years, were taken as the input value for determination of the wind load

The speed of wind in tornadoes, which could occur in the Czech Republic, corresponds to the extreme wind speed defined for frequency once in 10,000 years. Therefore, the extreme weather load can be considered to include the potential case of tornadoes. It can also be assumed that the potential effects of flying objects picked up by tornadoes are included in the requirement for resiliency of safety relevant buildings against the impact of external flying objects.

Heavy snowfall and ice

Snow load is expressed in terms of the water equivalent of snow, i.e. the height of the corresponding water column in mm. The input values for snow load and water precipitation are: 92 mm for a frequency of once in 100 years and 157 mm for a frequency of once in 10,000 years.

Ice cannot form in water management buildings with open water thanks to their function (transfer of heat) and the corresponding temperature conditions.

Because ice effects have formed in the past on the Vltava River near confluence with the Lužnice, the design included construction of the Hněvkovice and Kořensko reservoirs, which eliminate these effects so that the supply of raw water and disposal of waste water is secure even in case ice effects occur.

The water supply facility at the Hněvkovice reservoir was designed with the protection of the inlets into the pumping station remaining at about 8 m below the permanent level, which guarantees a problem-free supply of water in all states of operation.

Maximum and minimum temperatures

The effects of outdoor temperatures are assessed on the basis of the measurements of the outdoor air temperatures in the Temelín, Tábor and České Budějovice stations. As a conservative outcome value, the value determined from measurements in the Tábor station was used. The input values for the assessment of load due to temperatures were as follows: 39.0 °C as the maximum temperature of air and – 32.3 °C for the minimum air temperature (frequency of once in 100 years) and 45.6 °C as the maximum air temperature and -45.9 °C as the minimum air temperature (frequency of once in 10,000 years).

Consideration of the potential combination of weather conditions

SSC of 1st category seismic resilience were also assessed from the point of view of basic and extraordinary combinations of loads. The basic combinations of loads include permanent loads and accidental long-term and short-term loads. Extraordinary combinations of loads include permanent loads, accidental long-term and short-term loads and accidental extraordinary loads.

Permanent and long-term accident loads include, among others:

- o Snow load with a lower standardized value.
- o Temperature weather effects with a lower standardized value.
- o Short-term accidental loads include, among others:
- o Wind load

- o Snow load with full standardized value.
- o Temperature weather effects with full standardized value.

Accidental extraordinary loads include, among others:

- o Seismic effects
- o Extreme weather effects with a frequency of once in 10,000 years
- o Load from an external shock wave.

III.4.2 Evaluation of the safety margins

III.4.2.1 Estimation of the safety margin against extreme weather conditions

Safety margins for the effects of extreme conditions are given as the differences between the values of the design and extreme load. In case the design values for weather conditions were reached, it will be a subject for consideration whether to continue operation or shutdown the units, depending on the prognosis. With respect to the real weather conditions in the site of the Temelín NPP and resilience of the project, the following basic safety functions are ensured with a sufficient reserve:

a) Controlling reactivity

b) Transferring heat from nuclear fuel

c) Capturing ionizing radiation and radionuclides.

The reactor is shutdown by the clusters falling. This function is not influenced by extreme outside conditions.

All systems relevant for nuclear safety and necessary to transfer heat are located in enclosed (sturdy) construction objects or in underground structures, which prevents the operating media from freezing. In case of extremely low temperatures, it may be considered that water freezes in cooling towers or in the CBSS. The cooling towers could freeze if the units are shutdown; however, in this mode, they play no role in the fulfillment of safety functions.

If heated water is circulated via the CBSS, not even extremely low external temperatures can freeze the CBSS. The CBSS could only freeze if both pumps of one of the EWS systems were shut down for a long period of time. Should the shutdown period under extremely low outdoor temperatures be too long, the CBSS could freeze and the ESW pumps could not be restarted. However, the limits and conditions do not permit the deactivation of all ESW divisions in any mode. Even in unit shutdown mode, at least two ESW divisions must be in operation.

Barriers against radioactivity release –fuel cladding, pressure boundary of the primary circuit and containment – cannot be threatened by extreme weather events. The protective containment is a passive system, which cannot be threatened by extreme weather events and isolation of its piping and passageways is ensured by redundant separating components, which also cannot fail as a consequence of floods.

Media within the site of the Temelín NPP are supplied by piping systems laid down on pipe bridges. Because the pipe bridges are not secured against external events (earthquake, extreme weather conditions), they are not a part of the system ensuring safety functions.

Extreme wind would very likely lead to the full loss of the external power supply for both units (loss of the 400 kV and 110 kV external power supply), followed by a reduction in the output of both units to the level necessary for house consumption. In addition, extreme temperatures could cause a loss of power supply from the operating and reserve sources.

If the external power supply is lost, power is supplied from emergency power sources (DG), which are located in buildings made of concrete, resilient against extreme weather conditions. The operating reserves of diesel fuel in buildings protected against freezing temperatures is sufficient for several days of operation of the DG. Further supply of diesel fuel via pipes leading over technological bridges of the fuel management system of the NPP in the late stages of an accident is not ensured (freezing of the diesel fuel, damage due to extreme winds etc.). The further supply of diesel is ensured by tank trucks.

For maximum outside temperatures, calculations demonstrated that the maximum design value for the temperature of the ESW would be slightly exceeded only for a brief period of time in case of an LOCA. However, the probability of concurrence of extreme temperatures and a LOCA is very low. In other cases, the functioning of the ESW cooling system is not threatened.

Thermal calculations for minimum outside temperatures show that in case of extremely low temperatures, ice can form in the CBSS. However, this ice does not prevent operation of the CBSS. The slope of the walls in the ESW pool is big enough to permit the movement of ice (raising the level).

Conclusion on the adequacy of protection against extreme weather conditions

All objects within the 1st seismic category will withstand an extreme load caused by natural events. Analyses have proven resiliency against weather extremes for all SSC, which ensure or contribute to the fulfillment of basic safety functions.

Tab. 32: Values of derived extreme weather conditions for design level and extreme design level (with the exception of rainfall)

Event (weather conditions)	Design level (frequency of occurrence once in 100 years)	Extreme design load (frequency of occurrence once in 10,000 years)
Extreme wind [speed]	49 m/s	68 m/s 1)
Snow [equivalent water column]	92 mm	157 mm
Maximum temperature [peak value]	39.0 °C	45.6 °C
Minimum temperature [peak value]	-32.3 °C	-45.9 °C

1) Includes level F2 tornadoes

III.4.2.2 Measures that can be foreseen to increase the sturdiness of the plant against extreme weather conditions

The design and diversity of the SSC ensure sufficient resiliency and reserves in case of extreme weather events. The possible adverse effects of extreme natural events could lead to shutdown of units of the NPP, however they cannot pose a threat to the safety functions. The auxiliary functions could be influenced as a result of extreme natural events, e.g. in case the media in the bridge pipes freeze. Because bridge pipes are not secured as resilient against external events (earthquakes, extreme winds), these are not safety function systems.

The aim of the proposed measures is to further strengthen the defence-in-depth protection in case of extreme natural events. The opportunities to improve the defence-in-depth protection are listed in the following table. The table also contains an area requiring the execution of additional analyses, which were not available at the time the assessment was carried out.

Tab. 33: The opportunities to improve defence-in-depth against extreme weather conditions

Improvement opportunity	**Corrective measure**	**Time period** (Short-term / Medium-term II)	**Note**
Technical means	Alternative supply of diesel fuel from the tank for long-term operation of the DG	I	
Staff	Ensuring safety and operational staff in case of extreme events	I	
Analyses	Executing methodology for assessing external effects, verification of analyses carried out, possible technical measures	II	

III.5 Loss of electrical power and loss of the ultimate heat sink

Correct understanding of the following text requires familiarity with the contents of Chapter III.1 which describes the technological systems designed to fulfill the main and auxiliary safety functions within the Temelín NPP.

III.5.1 Loss of electrical power

Electrical systems in the Temelín NPP were designed to comply with the requirements of the mechanic and nuclear parts and respect the properties of the off-site power grid, especially with respect to the safety of operation of the Temelín NPP and production of electricity.

The safety of the Temelín NPP in case of a power loss was handled in the design by a high degree of diversification of the operating and reserve sources for house consumption, as well as by the redundancy and diversification of the secure power supply systems (SPSS), which contain own emergency sources and supply not only safety systems, but also safety related and otherwise important systems and components of both units. The power supply for house consumption is separate for each unit in order to prevent the spreading of electrical disturbances.

The operating source of power for house consumption of each unit is a pair of transformers connected to a branch of the power output from the 1,000 MW turbine generator. Thanks to generator switches, these house transformers can be supplied from two sources:

- 1000 MW turbo generator (at power operation of the unit)
- Kočín 400 kV switchyard low power operation).

This source is available under normal and abnormal operation, as well as in emergency situations, provided that either the link to the 400 kV power grid or power supply from the turbine generator remains intact. If the unit is disconnected from the 400 kV grid while the turbine generator is in operation, the supply for the house transformers is automatically switched to the turbine generator by regulating the output to the level of house consumption. A specific case is “off-grid” or “island” operation, when a failure in the electrical framework may lead to a disconnection of a certain part of the grid, while the turbine generator may remain connected to this “healthy” part of the system. The size of such an island cannot be determined in advance and it can differ greatly – from a large part of the framework to the extreme case of a minimum-size island, represented by operation of the TG for the unit’s house consumption.

House transformers supply the unit’s 6 kV switchboards of unsecured power supply. These switchboards normally supply the 6 kV switchboards of secure power supply, which then supplies the safety relevant systems.

The operating sources of power supply are not available during unit shutdown, e.g. during regular preventive maintenance of the 400 kV supply system.

A reserve source of power supply for each unit is a pair of reserve transformers supplied from the Kočín 110 kV switchyard using one 110 kV line (in total, 2 110 kV lines for both units). The 110 kV network in the Kočín switchyard can be supplied from several directions and hubs of the electrical grid. Reserve transformers are connected via reserve connections to 6 kV unit switchboards of unsecured power supply.

Reserve sources are used under normal and abnormal operation, as well as in emergency situations when the operating sources are partially or fully disabled. The reserve sources in both units are mutually redundant and can be switched using a manual connection switch. A reserve source is also able to replace the operating sources of one unit while partially supplying the other unit.

Emergency sources of power supply were designed for situations when both the operating and reserve sources fail. The emergency sources of power supply (DG, accumulator batteries) are safety sources (intended for one unit) with 3 x 100% redundancy and common (intended to be used by both units) with redundancy of 100% + 100%. Their operability does not depend on the availability of the operating or reserve sources.

Another option for emergency power supply is to use external diverse sources (hydro alternators in the HPS Lipno and hydro alternators in the small HPS Hněvkovice), which has been also tested in practice.

Power supply can be lost in one or both units of the Temelín NPP. Units operated at power have a higher design resiliency against loss of the electrical power supply (additional barriers of defence-in-depth), than during a shutdown for refueling. The least favorable case in terms of safety is loss of the power supply on both units simultaneously. In terms of possible configurations of the Temelín NPP, the the most controversial is the situation when one of the units is in emergency shutdown due to a loss of power supply and the other unit is in shutdown.

III.5.1.1 Loss of off-site power

Design provisions

Loss of the external power supply (e.g. in case of grid breakdown combined with the loss of the 400 kV and 110 kV switchyards) during at power operation does not automatically lead to activation of the emergency power supply.

If one unit is disconnected from an operating 400 kV power source during at power operation the operating 1,000 MW TG is capable by design of being regulated for house consumption and thus providing a power supply to all safety related systems. If this does not take place (unit shutdown, TG does not work, failed to regulate or failed), it is concluded that the unit's operating power sources failed. In such a case, the power supply for house consumption is automatically switched to reserve 110 kV power sources (large-scale automatic switch to reserve sources), the DG are started, the accumulators are being recharged in standard mode, and they provide a secure, uninterrupted power supply to the direct current power mains.

Only if the above automatic functions fail, is the LOOP signal generated as a result of loss of the power supply in the SPSS 6 kV switchboards. The SPSS 6 kV switchboards are automatically disconnected from the 6 kV switchboards of unsecured power supply and all three independent safety DG are started; after being connected to the SPSS 6 kV switchboards are they gradually put under load by the emergency load sequencer. These safety DG provide a power supply for the unit's safety systems, while each of the systems is sufficient for coping with transition events in case the external power supply is lost.

Together with the loss of voltage in the SPSS 6 kV switchboards in systems related to safety after they were disconnected from the 6 kV switchboards of unsecured power supply, common DG are gradually started and put under load. These DG provide a power supply for systems related to nuclear safety and systems for the safe rundown of turbine sets. In this situation, the accumulator batteries are recharged in standard mode and provide an uninterrupted power supply for the direct current power mains.

The operating or backup power supply can be incapable of operation for a long period of time low power modes of the unit, as a part of regular maintenance.

When the external power supply for the Temelín NPP is lost, none of the following basic safety functions is threatened:

a) Controlling reactivity

b) Transfer of heat from nuclear fuel

c) Capturing ionizing radiation and radionuclides.

Units of the Temelín NPP can, in case external power supply is lost, be maintained in a hot state for a long period of time, cooled to a cold state or safely maintained in shutdown mode. The power supply for all essential machine systems and I&C systems is ensured if at least one of the three safety DG in each unit, and at least one of the common DG, starts. However, to cool the unit to a cold state, it is sufficient if at least one of three safety DG on each unit starts.

If, in case of LOOP, the unit is under power, all main circulation pumps will be switched off and the signal for quick shutdown of the reactor will be sent. Residual heat is transferred from the active zone by natural circulation; steam is transferred from the SG via the station releasing steam into the atmosphere. Water is supplied into the SG via two auxiliary pumps from the water tank, which is resupplied by auxiliary condenser pumps either from the turbine condenser or from 2 x 800 m^3 reserve tanks of demi water. Alternatively, water can be supplied into the SG using emergency SG water supply pumps, which pump water from 3 x 500 m^3 tanks directly into the corresponding SG.

If a unit is shut down during LOOP, the heat from the active zone is transferred using a system for residual heat transfer. Each of the three cooling circuits includes a circulator pump and heat exchanger. The heat exchangers are cooled using essential service water. The pumps transferring heat from the active zone, as well as the ESW pumps, are supplied from the DG secure power supply systems.

Each section of the SFSP containing spent fuel is cooled using one cooling circuit. Each of the three cooling circuits includes a circulator pump and heat exchanger. The heat exchangers are cooled using essential service water. The pumps for SFSP cooling and the ESW pumps are also supplied from the DG secure power supply systems.

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

In accordance with the basic scheme of the machine and nuclear parts (3 redundant and independent divisions of safety systems) there are also 3 redundant and independent systems of secure power supply (3 x 100%). Each of these SPSS is a support system for safety systems of the corresponding division and its readiness to carry out safety functions is tested on a regular basis.

- Emergency sources of alternating current of the SPSS for safety systems are three independent (systemic) safety DG, which are connected to the corresponding connections to a 6 kV secure power supply.
- Emergency sources of direct current power supply are the accumulator batteries, which are permanently connected to the corresponding switchboards.

To ensure the necessary degree of redundancy, the SPSS of safety systems are independent and separated constructionally (independent rooms located in different parts of the reactor building, constructions with fire resistance for at least 90 minutes), electrically, as well as in terms of management by the control system. The SPSS are seismically resilient, i.e. comprised of seismically resilient systems; these are located in seismically resilient premises. Building constructions protect these premises against adverse effects in operation as well as in case of failures and natural events, which could take place inside and outside of the power plant. The cabling of each of the SPSS is led in independent cable routes separated from the cable routes of other SPSS (also with fire resistance for 90 minutes).

Besides the SPSS of safety systems, the project includes another two SPSS that supply parts of systems related to nuclear safety and systems irrelevant for nuclear safety, which ensure the general safety of persons and expensive system components. These SPSS are designed as two subsystems, which are mutually redundant according to the scheme 100 + 100%.

- The emergency sources of alternating current for the SPSS for systems related to safety are two diesel generators common for both units.
- The emergency sources of direct current power supply for the SPSS for systems related to safety are accumulator batteries, separate for each unit.

If voltage from operating or reserve sources is available for supplying the SPSS, the DG are kept as a hot reserve. In case the power supply from operating and reserve sources is lost, the power is supplied for the affected parts of the SPSS mains automatically by the corresponding DG.

The diesel generators have own diesel fuel tanks sufficient for operation under nominal load for at least 48 hours without refilling (realistically, even longer). These fuel tanks are also seismically resilient. Each common DG has a tank sufficient for about 12 hours of operation under 100% load (supplying both SPSS in both units).

Considering the real amount of fuel diesel in the reserve and standard tanks, the safety DG can operate under nominal load for at least 56 hours. With respect to the concept of safety systems redundancy (3 x 100%), it is possible to use the individual safety divisions one by one, and thus increase the period of power supply without diesel fuel resupply to about 7 days. All these time periods are estimated on the basis of the assumption of nominal load on the DG to supply about 5 MW. Realistically (considering the activities necessary for EOP, when only those parts of the system are running, which are currently necessary for safe operation of the unit), the actual load on the DG will be between 2.5 and 3 MW. This standard operating measure prolongs the period for which power supply is provided without resupplying the fuel diesel, by another 40% to about 10 days.

In addition to the tanks located near the DG, there is a diesel fuel management system with additional minimum volume of 300 m^3. Because the pumps of the diesel fuel management system are supplied from switchboards of the unsecured power supply, in case of the long-term loss of the external power supply it is necessary to resupply diesel fuel using mobile means of transport. When taking into consideration resupply using a mobile means of transport, it is possible to run the minimum necessary number of DG (one safety DG per unit and one common DG for both units) for at least another 3 days.

All auxiliary systems for the engine and generator DG (fuel intake, lubricating oil, internal cooling circuit, filling and initiating air) are autonomous and, while the DG is in operation, also independent of the outside energy supply. The systems there that could be influenced by long-term operation of the DG (clogging of the filters for lubricating oil) are redundant subsystems. One of these subsystems can be shut for maintenance while the DG is in operation. This system allows preventing failure of the DG as a result of auxiliary system failure. The operability of the DG and its auxiliary systems is checked on regular basis. The quality of the diesel fuel is checked monthly and it is kept in accordance with the corresponding requirements.

Battery capacity, duration and options to recharge batteries

The accumulator batteries are being recharged while the DG is in operation.
The capacity of the accumulator batteries of the SPSS safety systems is 3 x 1,600 Ah. For SPSS for systems related to safety, the capacity of the accumulator batteries is 2 x 2,000 Ah and 2 x 2 400 Ah.

In case the operating and reserve sources in the unit and connection to the DG are lost, the accumulator batteries are recharged in the ordinary mode and provide an uninterrupted power supply for the direct current power mains.

III.5.1.2 Loss of off-site power and loss of the ordinary backup AC power source

Design provisions

If the operating and backup power supply are lost along with emergency power supply for the unit (safety DG), the following sources are available to provide an AC power supply:

Internal sources:

a) Power supply from emergency supply sources of alternating current for the SPSS of systems related to safety (i.e. common DG – identical construction to the safety DG).

b) Supply from the other unit (with TG regulated for internal consumption).

c) External diverse sources (main strategy of the Temelín NPP for dealing with a loss of AC sources):

d) Power supply from the Lipno water hydro plant (2 x 60 MW) using dedicated lines. In case of a power grid breakdown, power can be fed to the Temelín NPP from the HPS Lipno, which can be powered up without an external power supply (from the dark). In case of a power grid breakdown, it can be powered up and, after the route is set by the control centre, deliver voltage for house consumption in the Temelín NPP. The time necessary for bringing voltage from the HPS Lipnoto the Temelín NPP is about 30 minutes, and the feasibility of this variant was confirmed by a test (verification of the organizational measures for coping with an SBO, functionality of the TSPP, functionality of means of communication, roles and procedures of key persons in case of an SBO).

e) Power supply from the small HPS Hněvkovice – source with a small output (from 2 x 2.2 MW to 2 x 4.8 MW, depending on the water flow).

f) Voltage can be brought to the Temelín NPP via the Kočín 110 kV switchyard using the 110 kV line for backup power supply.

The ability to use both external diverse power sources of the NPP (HPS Lipno and hydro alternators in the small HPS Hněvkovice) is also ensured in case the power grid breaks down, thanks to their ability to be launched “from the dark”. Energy from both sources would be brought to the Temelín NPP via the Kočín 110 kV switchyard along the line of the 110 kV backup power supply (in case the HPS Lipno is used for setting up the route via the 110 kV distribution network.).

Other sources of alternating current power sources, which are by design not intended to supply safety systems in case of an SBO, are available for the Temelín NPP.

- DG for supplying the lubricating pumps of the turbine (output of 200 kW).
- DG for the data centre (output of 1 MW).

Works on recovering the power supply for the safety systems from on-site and off-site sources can be carried out in parallel.

Although the option of connecting these sources to the existing power supply mains is not included in the project, nor has it been tested, their output is sufficient for the long-term recharging of the accumulator batteries.

III.5.1.3 Loss of off-site power and loss of the ordinary backup AC power sources, and loss of permanently installed diverse backup AC power sources

The full loss of all sources of a unit's AC power sources is highly beyond the design basis and unlikely accident. It can only take place if all the following levels of defence-in-depth for power supply failed at the same time:

(1) Off-site operating power sources – normal power supply from a 400 kV switchyard.

(2) On-site operating power sources – failure to regulate the turbine generator for house consumption.

(3) Off-site backup power sources – backup power source from a 110 kV switchyard.

(4) On-site backup power sources – power supply from a 110 kV switchyard of the other unit.

(5) All three redundant emergency AC power sources for the SPSS of safety systems (safety DG) in both units.

(6) Both emergency AC power sources for the SPSS for safety related systems (common DG).

(7) Diverse off-site AC power sources (hydro alternators from the HPS Lipnoand small HPS Hněvkovice).

Battery capacity, duration and possibilities of recharging batteries in this situation

In this mode, the accumulator batteries would not be recharged (the recharge period is in hours, depending on the current load). If the power supply is not recovered and the heat is not transferred from the I&C systems over time, the ability to communicate the values of key parameters, control circuits, emergency lighting etc. would gradually fail (due to discharged batteries).

The discharge time for the batteries of safety systems depends on the load over time and it is expected to be a few hours.

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

The location has available on-site alternative or mobile AC power sources, which by design are intended for dealing with a long-term SBO; however, there are also off-site sources, whose availability and feasibility for solving an SBO has been verified and tested.

Alternative means of transporting media include the mobile equipment of the on site fire brigade. However, employing this equipment for technological purposes has not been described. It is necessary to verify the capacity and readiness of the connections, which would allow connecting this equipment to systems ensuring basic safety functions.

Competence of the shift staff to make the necessary electrical connections

In order to ensure the fulfillment of safety functions (for design basis and beyond design basis scenarios) and recover the power supply in case of an SBO, corresponding procedures have been executed for the phase preceding damage to the reactor core in the EOPS and for situations involving damaged fuel in the reactor core in the SAMG. There are also procedures elaborated for recovering the power supply in case of an SBO from on-site sources, as well as from the HPS Lipno as an off-site power supply.

Returning the unit to a safe state after an SBO before the fuel in the core is damaged has two phases. Activities are first carried out according to the procedure for full loss of the 6 kV secure power supply. The activities described in these procedures end at the moment the power supply in at least one division of the safety systems has been recovered. To recover the power supply, one of two related procedures for subsequent steps is to be used, depending on the state of the unit. If no significant changes or disruption of the unit's parameters (temperature and pressure in the I.C, levels in SG etc.) have taken place and there is no request for activating emergency resupply for the I.C, the procedure is to be used which stabilizes the unit's parameters using the available systems. If the state or the unit's parameters change significantly before the power supply is recovered and the conditions for launching emergency supply for the I.C are fulfilled, a different procedure is to be used – a procedure which describes the steps using systems of emergency supply of the I.C. The steps necessary for bringing power from the corresponding selected power source are described in separate procedures.

The staff in the Temelín NPP is sufficiently qualified and trained in using EOPs and SAMGs, as well as in carrying out the activities necessary for ensuring power supply from off-site and on-site sources in case of an SBO. No deficiencies in terms of the staff numbers necessary for mitigating the consequences of an SBO have been identified in the IOER (internal organization of emergency response) or SOER (standby organization of emergency response).

In case of long-term loss of all AC sources (SBO), the consequent loss of lighting (until the accumulator batteries, run out emergency lighting would be available) would make moving inside the NPP much more difficult for the staff. This would also increase the time necessary for carrying out the necessary steps. In case of an SBO in both units, the current staff could be overloaded trying to recover the power supply.

With a shutdown unit it would be necessary to close the containment at the right time. Although there are technical means for isolating the containment, at this time no specific procedures for isolating the containment in case of an SBO (with deactivated units) have been executed.

In case of a full loss of the power supply, the first priority is to carry out steps to prevent the loss of the primary and secondary coolant. Loss of the primary coolant is prevented by keeping all discharge routes from the I.C closed. Loss of the secondary coolant can be prevented by maintaining high pressure in the SG (regulated discharge of steam from SG into the atmosphere), i.e. by closing the outlet routes. The heat must be transferred from the I.C in a way so that the pressure in the I.C does not exceed the limit value, which would lead to opening of the pressuriser relieve valve. All these steps can be carried out using systems supplied from batteries. If leaks from the primary and secondary circuits are prevented, it is possible to keep the unit in a hot state for some time without posing an immediate threat to the safety functions.

The Instructions of TSC address the philosophy of a possible relief for the DC power supply, which would allow prolonging the discharge period for the accumulators; what needs to be done now is to execute detailed procedures for relieving the DC power supply and batteries of the safety related systems.

According to the conclusions from the inspection of discharging carried out for the batteries of the safety systems, considering a partial reduction of the load (after 30 minutes, 30% reduction of the load), the battery is capable of carrying load for over 4 hours. With respect to the concept of safety system backups with 3 x 100% redundancy, it is possible to gradually employ individual safety divisions and thus increase the period during which this power supply is ensured, to 12 hours.

Further prolongation of the discharge period for the batteries of the safety systems can be achieved by employing the batteries of the safety related systems, whose capacity is comparable to the capacity of the batteries for the safety systems. To cope with an SBO, it is necessary to know the values of the key parameters of operation. The values of the safety relevant variables are communicated via the PAMS (post-accident monitoring system). The corresponding I&C, which communicates the values of the parameters, as well as the PAMS, are supplied from accumulator batteries. The parameters could be lost if the systems supplying the I&C fail or if the temperature in the I&C rooms increases. The shortest possible period during which the values of the key operating parameters are communicated, is always longer than the period sufficient for endangering the transfer of heat from the core.

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

The most limiting factor in case of an SBO (transfer of heat from the I.C, transfer of heat from the spent fuel pools, loss of cooling in the I&C rooms, discharge period of the accumulator batteries) is the time until the fuel in the core is damaged. Another aspect limiting the time a unit can stay in SBO mode, is the discharge time of batteries. Until the batteries are discharged, the power supply for key valves, I&C for communicating the values of key parameters, control circuits, emergency lighting etc. is preserved.

In case of an SBO, the ultimate heat sink for the transfer of heat from systems supplied by accumulator batteries is unavailable, which poses a threat to the I&C overseeing the safety systems. In case cooling of these systems is not recovered, the proper functioning of the I&C could be compromised even if a long-term power supply is provided.

When unit is run at power or in a hot state, an SBO would lead to a loss of power for the SG and thus a decrease in the amount of water available for secondary parts of the SG. The pressure in the SG is regulated by releasing steam into the atmosphere. This would lead to gradual uncovering of the tubes of the heat exchanger in the SG and thus to a reduction in the effective heat exchange surface for transferring heat from the I.C. This situation would lead to a loss of the secondary heat transfer system. From the moment the SG are unable to transfer all the residual heat produced in the core, the temperature of the coolant in the I.C would increase. Due to thermal expansion of the coolant, this would lead to an increased level volume compensator and consequently, to increased pressure in the I.C.

Until the power supply is recovered after an SBO, the water supply in the SG guarantees the transfer of heat from the core via the SG into the atmosphere for several hours. The limiting condition in case of an SBO is the time until the fuel in the core would overheat. In the worst-case scenario, the temperature could reach 650 ^{o}C at the outlet from the core within hours after an SBO. The same time period would also be available to recover the power supply in case the heat transfer function is lost in the shut down unit and the level of coolant in the core dropped, while it is still possible to let water gradually pour into the core by gravity effect. If heat cannot be transferred from the SFSP, the stored fuel would not start overheating for tens of hours after the SBO.

From the analyses of SBO scenarios, including the failure of heat transfer from the I.C using the SG, it follows that without carrying out the alternative activities described in the EOPs, there is very short time reserve for restoring the heat transfer from the I.C. The temperature of 650 °C at the outlet from the core, which is the limiting value in terms of fuel damage, would, in the worst-case scenario, be reached after 2.5 – 3.5 hours after the SBO.

When in deactivated state with the level along the axis of the cold nozzles an SBO combined with no alternative steps being carried out (as described in the EOPs – gradual gravitational pouring of water from the hydro accumulator into the core) could bring the coolant to boiling in the worst-case scenario (the level in the reactor declining immediately after the core was deactivated and cooling to a cold state) within 10 minutes after the core cooling system is lost. The fuel would overheat within 30 minutes.

To ensure the transfer of heat, it is necessary to restore power supply for at least one safety connection within this time period. This would prevent uncovering and possible damage to fuel in the early stages of an accident.

As a consequence of a power supply loss, the cooling of spent nuclear fuel would be interrupted and the water in the SFSP would warm up. The trend of temperature increase in the SFSP after cooling stops depends on the initial conditions (time since the spent fuel was removed from the reactor, amount of fuel in the SFSP, etc.). Even under maximum temperature load on the SFSP, loss of the cooling function in the SFSP will not lead immediately to damage to the stored spent fuel; it would take at least tens of hours.

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

Power supply sources in the Temelín NPP provide a sufficient design basis sturdiness and degree of safety in case of power supply failure. By design, the operating and backup sources for house consumption have a high degree of independence and the systems of secure power supply are sufficiently redundant. These sources supply safety relevant systems and components and have own emergency sources (DG and accumulator batteries). Power for house consumption is supplied per unit, which prevents failures from spreading between units.

When operating at power, the design basis resilience against loss of power supply is higher (additional defence-in-depth barriers) than during a shutdown for refueling. In terms of safety, the worst-case scenario is a loss of power in both units simultaneously.

There are a total of 8 emergency sources of alternating current in the location (3 safety DG for each unit and 2 common DG for both units). When the external power supply is lost the Temelín NPP, the units can be kept in a safe state in the long run, cooled to a cold state or kept safely in shutdown mode (power is supplied to all necessary machine systems and the I&C), provided that at least one of these DG starts in each unit. For each DG there is enough diesel fuel for 2 – 3 days without the need to refuel. Additional volume of diesel fuel for the long-term operation of the DG is available on site.

If the external power supply is lost and the TG is not regulated for house consumption safety systems, safety related systems and systems for safe rundown of the turbine sets are supplied by emergency sources of AC power supply (DG) and by emergency sources of uninterrupted DC power sources (accumulator batteries). The batteries are being recharged constantly, as long as the corresponding DG is in operation. Long-term operation using emergency sources would require the additional supply of fuel diesel using mobile means of transport.

In case of a full loss of the AC power supply (SBO) and if all the following levels of defence-in-depth fail at the same time:

(1) External operating sources – normal power supply from a 400 kV switchyard,

(2) Internal operating sources – failure to regulate the turbine generator for house consumption,

(3) External backup sources – backup power supply from a 110 kV switchyard,

(4) Internal backup sources –power supply from a 110 kV switchyard in the other unit,

(5) All three redundant emergency sources of AC power supply for the SPSS of safety systems (safety DG) in both units,

(6) Both emergency sources of AC power supply for the SPSS of safety related systems (common DG),

(7) Diverse external sources of AC power supply (hydro alternators in the HPS Lipno and hydro alternators in the small HPS Hněvkovice),

the only sources supplying safety systems and safety related systems are emergency sources of uninterrupted DC power supply (accumulator batteries). If the corresponding DG does not run, the accumulator batteries are not being recharged and their discharge period takes hours, depending on the load. Considerable prolongation of the discharge time is possible by a controlled reduction of the load, by gradual one-by-one use of individual divisions and by using the accumulator batteries of safety related systems, which have a high capacity. Alternatively, other sources of AC power supply available in the Temelín NPP could be used for the long-term recharging of accumulator batteries.

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

In case of a full loss of AC power supply (SBO), emergency sources of uninterrupted power supply (accumulator batteries) are the only source of power supply for safety systems and safety related systems. In case of an SBO, the long-run fulfillment of the safety functions depends on the ability to restore AC power supply, which, according to the current project, can be done in several ways.

Although an SBO can only occur if several levels of defence-in-depth in the electrical part of the NPP fail simultaneously, the potential consequences of an SBO require that further measures are proposed to increase the already very high sturdiness of the project in terms of power supply for house consumption from safety systems, including the possibility of adding and testing alternative sources to the existing power mains.

The goal of the proposed measures is to strengthen the levels of defence-in-depth in case of initiating events beyond the design basis (earthquake, floods, extreme conditions, human factor, etc.), which could lead to a loss of safety functions during an SBO:

- Propose and implement alternative sources of AC power supply for the systems providing cooling and heat transfer from the core and SFSP and the method of their connection to the existing power mains.
- Propose and implement diversified means of protection for systems providing cooling and heat transfer from the core and SFSP, and the method of their connection to the existing systems.
- Propose and implement alternative means ensuring DC power supply and cooling of the I&C systems necessary for monitoring the situation and controlling selected components.
- Propose and implement alternative means for activities and functional communication (internal and external) of the staff.

Specific opportunities for improving defence-in-depth protection can be found in the table 34. The table also contains areas that require additional analyses, because these were not available at the time this assessment was performed.

III.5.2 Loss of the ultimate heat sink

III.5.2.1 Design provisions to prevent the loss of the primary UHS

The ultimate heat sink for both units of the Temelín NPP is the atmosphere. Unused heat from operation at power of the units, i.e. residual heat from the core after shutdown, can be led to the ultimate heat sink in several ways:

- Transfer of heat via the TG condensation system into the circulation cooling water and via cooling towers into the atmosphere – under ordinary and extraordinary at power operation, activation as well as shutdown of the TG and under emergency conditions after the reactor was shut down, provided that operating or backup sources of power supply are available.
- Transfer of residual heat from the core and components of safety systems using the essential service water system into the cooling towers and from there into the

atmosphere – under ordinary and extraordinary operation and in emergencies, with the possibility of bringing the reactor into a cold state.

If operating means of heat transfer into the ultimate heat sink are unavailable, it is possible to employ alternative methods of heat transfer:

- Direct transfer of heat by releasing steam from the SG into the atmosphere while adding supply water – in abnormal or emergency operation; this variant allows long-term transfer of heat from the core, but does not allow cooling the reactor to a cold state (cooling to approximately 110 ^{o}C).
- Alternative “feed and bleed” method (controlled discharge of coolant from the I.C into the containment, transfer of heat via ECCS exchangers into the ESW and supply of cooled coolant using pumps for emergency resupply into the I.C – only in emergency conditions when it is impossible to use a secondary transfer of heat.

After reactor shutdown with the unit kept in the hot state, or in the first phase of unit cooling heat is transferred from the I.C via a secondary SG; supply water flows into the SG (system of ordinary or emergency supply) and steam is led out of the SG into the SG condenser or into the atmosphere. The transfer of heat from the SG condensers to the circulation cooling water is not assessed further, because it is not essential that it is available (non safety related system).

The necessary flow into each SG can be provided by a backup power supply system for the SG (redundancy 2 x 100%), as well as by the system of emergency power supply for the SG (redundancy 3 x 100%). The residual output of the core can be transferred by any of the four stations releasing heat into the atmosphere, which are inseparable parts of the corresponding SG. Heat can be transferred into the atmosphere from the I.C down to the temperature of 110 ^{o}C; after that, the system is inefficient.

Essential service water system can be used to cool the I.C into a cold state, to transfer heat from spent fuel in the SFSP and to transfer heat from safety system appliances and from systems related to nuclear safety. The ESW system transfers heat via the CBSS into the atmosphere, as the ultimate heat sink. All three ESW systems are in operation at the same time (redundancy 3 x 100%). The heat is transferred from each ESW system into a separate CBSS, where it is discharged into the atmosphere by the vaporization of water from the water surface and from water sprayed by jets. Each SPSS consists of two functionally independent halves; when one half of SPSS is in operation, the other can be empty or available as backup.

An alternative system for the transfer of heat from spent fuel located in in the SFSP is the containment spray system, which resupplies water into the SFSP via a dedicated route and discharges heat by vaporization into the containment.

The ultimate heat sink used by the Temelín NPP cannot be lost. The transfer of heat from the core, the SFSP and safety systems into the atmosphere as the ultimate heat sink is based on the passive physical principle of heat conduction from the medium into the atmosphere. The ultimate heat sink can, therefore, be considered lost only when the ability to transfer heat is lost, i.e. when systems ensuring the flow of media for heat transfer between the heat sources and the atmosphere fail. For the purposes of assessing the possible loss of the ultimate heat sink, it is, therefore, possible to consider the loss of the systems transferring heat via the secondary circuit and loss of ESW systems. Because the transfer of heat via the SG can only be used to reduce parameters in the I.C to values permitting activation of the system for the transfer of residual heat, which then passes heat on to the ESW systems, for the purposes of this assessment the loss of ultimate heat sink means, in particular, the loss of the ESW system’s ability to transfer heat from the core, SFSP and safety systems into the atmosphere.

This approach is also justified because the transfer of heat into the atmosphere directly from the SG using the secondary “feed and bleed” method can be ensured by a combination of several diversified (from a system design point of view) and redundant (from a configuration point of view) systems. Systems participating in heat transfer are available also in case only emergency sources of power supply are available. Moreover, if the heat transfer function via the secondary circuit is lost, the EOPs describe a method of emergency heat transfer using the primary “feed and bleed” method.

Thanks to the fact that systems transferring residual heat are independent and separate for each unit, the potential loss of the ability to transfer heat is assessed separately for each unit. Loss of the ultimate heat sink poses a smaller threat for the transfer of heat from the core in modes during which the reactor is sealed (all modes, except for shutdown for refueling), because it involves the possibility to transfer heat via the SG. The shutdown mode, during which the reactor is not sealed and the transfer of heat via the SG is no long efficient, creates the same risk for the fuel stored in the reactor as for the fuel stored in the storage pools.

Water for technological purposes within the Temelín NPP is supplied from the Hněvkovice reservoir, right next to the dam on the left bank. The pumping station was designed to supply 1.3-4.16 $m^3.s^{-1}$ with the maximum amount requiring the operation of 1-4 pumps (capacity sufficient for the originally planned output of 4 x 1,000 MW). The other two pumps are 50% backup. The pumping station, therefore, contains 6 vertical pumping systems.

Water is transported into 2 x 15,000 m^3 water towers within the Temelín NPP compound by two pressure pipings made using DN 1600 mm steel pipes leading underground for about 6.2 km. The capacity of both pipings is 4.16 $m^3.s^{-1}$; in case of a breakdown of one piping, the other is capable of carrying a guaranteed amount of 3.4 $m^3.s^{-1}$ with 4 pumps running. The pumping station is supplied from double 110 kV mains from the Kočín 400/110 kV switchyard into the 110/6 kV switching and transformer station within the premises of the pumping station. Information about operating conditions and failures in the pumping station is transferred to the water management centre in the Temelín NPP via a communication cable.

III.5.2.2 Loss of the primary ultimate heat sink

Availability of an alternate heat sink

Calculations demonstrate that one CBSS is in the worst-case scenario able to transfer all the heat from the unit with shut down reactor. This can be sustained in the long run without resupplying. The temperature of the ESW in such case would not significantly exceed the maximum design value. The worst-case scenario is a situation when one unit experiences a LOCA and the other unit is being shut down, i.e. heat transfer into the ESW is peaking. Because there are three redundant ESW systems, it can be demonstrated that heat can be transferred into the ultimate heat sink without an additional supply of water for at least 30 days, provided that all safety divisions are used one by one, or that water from the CBSS of the non-functional ESW systems will be moved using a mobile means of transport into the SPSS of the operational ESW system.

Thanks to the 3 x 100% redundancy of the ESW systems and additional internal redundancy of pumps in each 100 + 100% ESW division, the loss of the heat-transferring capacity from the sources into the SPSS would require the failure of all ESW pumps (6 pumps in total). With respect to the fact that the systems and pumps are in different locations and that they have separate sources of power supply, as well as other auxiliary systems, the concurrent failure of all ESW pumps is extremely unlikely. With at least one pump in one division of the ESW system, it is possible to carry out basic safety functions. The only realistic events leading to failure of all ESW pumps are external floods and an SBO.

However, even in case the ESW is completely lost, the heat is transferred in the hot state from the core by the standard operating systems, which do not depend on the ESW system –

water is supplied into the SG using auxiliary pumps and steam is led into condensators or the atmosphere.

Non-technological means that can be employed in case the ultimate heat sink is lost include LFRU equipment. Until now, these means have not been considered for reducing the consequences of technological failures. Besides this equipment, no other alternative or mobile sources are available in the Temelín NPP that could help circulate (i.e. transfer) heat from the ESW system and thus contribute to solving a situation involving the loss of the ultimate heat sink.

Calculations demonstrate that one SPSS is capable of transferring all heat from both units for 12.5 days without an additional supply of water. To fulfill the requirements on the security of heat transfer for at least 30 days, it is necessary to move the water reserve from the SPSS of non-functional ESW systems using mobile means of transport into the SPSS of the functional ESW system. An analysis of the feasibility of mobile fire equipment implies that these means are sufficient for moving water between the SPSS.

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

Loss of heat transfer into the ultimate heat sink implies that it is impossible to cool the unit. With a stable or increasing temperature in the I.C, no positive reactivity is being added and, therefore, there is no need to compensate for additional positive reactivity. As long as the units remain in a hot state, the reactivity control function is preserved.

Because the functioning of safety systems supplying the I.C in emergencies, the containment spray system, emergency water supply to the SG, etc. depends on the functioning of the ESW system, they cannot be used directly for solving the situation. The transfer of heat from the core should be provided by systems of ordinary operation, which do not depend on the functionality of the ESW system – the supply of the SG system using auxiliary pumps, the transfer of steam into the condenser or the atmosphere. In a hot state, the secondary “feed and bleed” mode allows the long-term transfer of heat from the I.C.

Shutdown (opened reactor) is a completely different situation; the transfer of heat from the core depends on the functioning of the ESW. The consequence of the loss of the ESW is increased temperature in the core. In such a case, all wet transport pools can be filled. Without heat transfer, the temperature in the wet transport pools increases up to the saturation limit. Assuming vaporization is compensated by supplying more water, this mode allows long-term heat transfer.

In case the ESW fails, the situation for heat transfer from the SFSP is the same as in the case of an SBO, i.e. the cooling of spent fuel is interrupted and the water in the SFSP begins to warm up. The temperature trend in the SFSP after cooling is interrupted depends on the initial conditions (time since the spent fuel was moved from the reactor, amount of fuel in the SFSP, etc.). Even with maximum residual heat output, there is no immediate threat to the integrity of the spent fuel immediately after the heat transfer system is lost – it could be damaged in tens of hours, i.e. in a late phase of an accident.

Under normal and abnormal conditions, heat from the containment is transferred using air ventilation systems cooled by the ESW. In case of a temperature increase, the cooled water system can be used to transfer heat from the containment. In case the ESW and cooled water system are disabled for an extended period of time, the heat transfer from the containment stops and the temperature in the containment will begin to rise. However, cool air from outside will continue to circulate and underpressure will be maintained by air suction systems.

With respect to the construction and thermal and pressure sturdiness, the integrity of a closed containment would not be threatened until the late phase of an accident. Because there are no procedures for early closing of the containment, an open containment (especially states with an open reactor) could lead to a leak of radioactive substances from the coolant responsible for maintaining the temperature at the limit of saturation outside of the containment.

To secure safety functions, it is necessary that the entire I&C system is functional and that the values of the key unit parameters are known. After losing the ESW, the corresponding I&C, as well as the PAMS itself, will be affected by the increased temperature in the I&C rooms. The heat from the I&C rooms of the safety systems can be transferred alternately using the non-essential service water system. This possibility is described and commonly used during planned deactivation of the ESW and it increases the resiliency of the safety functions in case of the ESW failure.

III.5.2.3 Loss of the primary ultimate heat sink and the alternate heat sink

External actions foreseen to prevent fuel degradation

An alternative means of transport for the media is the mobile equipment of the LFRU. Using this equipment has not been described; therefore, it is necessary to verify the capacity of this equipment and readiness of connection spots, which would allow connecting this equipment to the systems providing basic safety functions.

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

In case a unit is operated under power or in a hot state and the ultimate heat sink is lost, it is possible to transfer heat from the core by ordinary operating systems, which do not depend on the ESW system – supplying water to the SG using pumps, moving steam into the condenser or releasing it into the atmosphere. In a hot state, the secondary “feed and bleed” mode allows the long-term transfer of heat from the I.C.

The situation is different during unit shutdown (with the reactor open), with the heat transfer from the core depending on the operation of the ESW. Loss of the ESW in such a case leads to temperature increase in the core. In such a case, the wet transport pools can be filled. Without heat transfer, the temperature in the wet transport pools increases up to the saturation limit. Assuming that vaporization is compensated by adding water, this state allows long-term heat transfer.

From the long-term perspective, it is necessary to restore operation of the ESW systems in at least one safety division, which would allow cooling the unit to a cold state.

Another important aspect that significantly influences the time period necessary for reaching saturation temperature in the SFSP is the water level. If the water level in the SFSP drops below 754 cm, circulation via the SFSP cooling system is lost, and if the water level drops below 550 cm, the heads of the stored fuel sets will be uncovered. Following the loss of the last functional redundancy for the SFSP heat transfer system, and after reaching the saturation temperature, the system experiences a significant “cliff edge” effect, which requires the further transfer of heat by boiling the coolant in the SFSP and releasing the vapor into the containment.

If the SFSP produces the maximum residual output (the entire core is transported and the rest of the SFSP is filled with spent fuel from previous campaigns), the minimum time until the saturation temperature is reached is 30 hours. The volume of water in the SFSP while storing fuel in each of Sections 01 and 03 is about 223 m^3 and in Section 02 it is about 104 m^3 (in the refueling mode, the volume doubles). With respect to the above volumes in individual sections of the SFSP, the reserve of coolant in the reservoir prolongs the time until saturation to about 60 hours, and the reserve of coolant in the reserve tanks for refueling extends the time until saturation further still, to about 120 hours.

As a consequence of losing the ultimate heat sink, the cooling of spent nuclear fuel would be interrupted and the water in the SFSP would start warming up. The trend of temperature increase in the SFSP after cooling stops depends on the initial conditions (time since the spent fuel was removed from the reactor, amount of fuel in the SFSP, etc.). Even under the maximum temperature load on the SFSP, the loss of the cooling function in the SFSP will not lead immediately to degradation of the stored spent fuel; it would take at least tens of hours.

Loss of the UHS would threaten the functioning of I&C for the safety systems because it would not be possible to transfer heat from the components supplied by the accumulator batteries. In case cooling for these components is not restored, the proper functioning of the I&C could be gradually compromised.

III.5.2.4 Conclusion on the adequacy of protection against loss of the ultimate heat sink

In the case of the Temelín NPP, the ultimate heat sink is the surrounding atmosphere. Unused heat from at power unit or residual heat after reactor is shut down can be transferred into the ultimate heat sink – the atmosphere - in several ways. Heat is transferred between heat sources from safety relevant systems and the atmosphere by the ESW system via the CBSS.

Available within the Temelín NPP is a water supply in the CBSS sufficient for 30 days of operation for the ESW system transferring residual heat from shutdown reactors without an external supply of water. Each unit contains a total of 6 ESW system pumps. Thanks to separation of the systems and pumps and independence of the power supply and other auxiliary systems, the concurrent non-functionally of all ESW system pumps is extremely unlikely. Even just one functional pump in one division of the ESW system can fulfill the basic safety functions.

Because there are several operating methods of transferring heat and, if these are not available, also several alternative methods in the extremely unlikely event that the ESW system's ability to transfer heat from the core, SFSP and safety systems into the atmosphere is compromised, it would still be possible to keep the unit in a hot or semi-hot state for an indefinite period of time by transferring heat into the atmosphere via the SG, which are independent of the ESW system. This possibility provides sufficient time to prepare alternative methods of heat transfer. Loss of the ESW is always connected with the inability to cool the unit to a cold state and maintain it in a cold state for a long period of time.

If neither operating nor alternative methods of heat transfer are available, the potential consequences of the absence of the heat transfer function could include:

- Damage to the fuel in the core and to spent fuel stored in the SFSP due to the non-existence of alternative methods of heat transfer from the core, SFSP and components cooled using the ESW.
- Loss of the cooling of emergency sources of AC power supply in case of LOOP can lead to an SBO.
- In case of disrupted isolation of the containment during shutdown, radioactive substances may leak into the surrounding environment.
- Loss of the ability to control systems and components and communicate the values of key parameters due to a failure of the I&C caused by the inability to transfer heat from the I&C components.

III.5.2.5 Measures that can be foreseen to increase the sturdiness of the plant in case of loss of the ultimate heat sink

Although full loss of the capacity to transfer heat into the ultimate heat sink would require combined failures within defence-in-depth levels, the potential consequences of such a situation require additional measures that could help increase the already very high sturdiness of the project in terms of the capacity to transfer heat into the atmosphere as the ultimate heat sink.

The aim of the proposed measures is to strengthen the levels of defence-in-depth in case of initiating events beyond the design basis (earthquake, floods, extreme conditions, human factor, etc.), which could lead to the loss of safety functions during an SBO:

1. Propose and implement diversified methods of cooling and heat transfer from the core and the SFSP, including the possibility of their connection to existing systems.
2. Propose and implement alternative methods for securing the cooling of the I&C necessary for monitoring the situation and controlling selected components.
3. Describe the use of alternative and diversified means (proposed according to Points 1 and 2) – "emergency plans" (EDMG), with the aim of securing cooling and heat transfer from the core and the SFSP.

Opportunities to improve the defence-in-depth protection in case of events that may result in the loss of capacity to carry out safety functions can be found in the table 34. The table contains areas that require the execution of additional analyses, because at the time this assessment was carried out these were not available.

III.5.3 Loss of the primary ultimate heat sink, combined with a station blackout

III.5.3.1 Time of autonomy of the site before the loss of the normal reactor core cooling condition

Loss of the primary ultimate heat sink, combined with a station blackout

There is only one ultimate heat sink within the Temelín NPP for the transfer of heat from the safety systems, and that is the surrounding atmosphere. The means of transferring residual heat from the core, from spent fuel stored in the SFSP and from the components of safety systems into the ultimate heat sink is the ESW system.

During SBO events, the ESW pumps have no power supply. Because it is the ESW system that transfers heat into the atmosphere, an SBO implies the loss of the forced heat transfer from the I.C and SFSP into the atmosphere. An SBO event will automatically lead to a loss of the ultimate heat sink in the given unit, as a consequence of the loss of the power supply for the pumps of the ESW system.

If the ultimate heat sink is lost and, at the same time, the power supply from the operating and backup sources are lost, the failure of the DG cooling will lead to an SBO situation within the unit. The reason for this is the dependency between the DG and the ESW – the failure of one means the failure of both.

The transfer of heat from the core via the secondary circuit (SG) can be used for transferring residual heat only in hot or semi-hot states of the units and only until the water supply in the SG has been consumed. However, there is no backup for heat transfer from spent fuel stored in the SFSP.

The above facts imply that the operability of the ESW system transferring heat into the ultimate heat sink and the operability of emergency power supply sources are interconnected.

Loss of the ultimate heat sink due to an SBO

The pumps of the ESW system, which provide the transfer of heat from the sources into the ultimate heat sink, are supplied from a secure power source. In case of an SBO, the ESW system is always lost. In such a case, there is a possibility to transfer heat from the core using the water supply in the SG directly into the atmosphere so that the ultimate heat sink is, in fact, not lost immediately. The capacity to transfer heat from spent fuel in the SFSP would be lost in a later phase of the accident.

Loss of the ESW in case of an SBO would limit the time period during which the values of important parameters of the unit and the NPP are available. Heat losses from the I&C system supplied from accumulator batteries without a cooling function, which is lost due to unavailability of the ESW system, will lead to a temperature increase in the I&C rooms and subsequent loss of the corresponding I&C systems.

<u>SBO as a consequence of a loss of the ultimate heat sink</u>

The loss of the ultimate heat sink itself will not have an impact on the power supply for the unit's house consumption if there is a power supply from operating or backup sources. In case the external power supply is lost and the turbo generator is not regulated for house consumption while the ESW is non-functional, emergency AC power sources will be activated (safety DG); however, after these are connected to the power mains and put under load, they will gradually fail due to overheating caused by a loss of ESW cooling.

The power supply for safety systems will be supplied only by accumulator batteries. Because the heat from I&C will not be transferred, the temperature in the rooms will increase and eventually, the I&C will be lost and thus, the ability to communicate important parameters will be reduced.

III.5.3.2 External actions foreseen to prevent fuel degradation

Alternative means of transporting media include the mobile equipment of the LFRU. The LFRU has at its disposal 4 tank trucks with pumps, 1 combined fire truck and 3 trailer fire pumps with a total nominal output of 280 l/s. However, employing this equipment for technological purposes has not been described. It is necessary to verify the capacity and readiness of the connections that would allow connecting this equipment to systems ensuring basic safety functions.

III.5.3.3 Measures that can be foreseen to increase the sturdiness of the plant in case of the loss of the primary ultimate heat sink, combined with a station blackout

The power supply from emergency sources and transfer of heat into the ultimate heat sink are closely related; the loss of one function could influence the performance of the other, and vice versa. Although the loss of ability to use both these functions would require multiple failures at the defence-in-depth level, the consequences of such a situation require that further measures be proposed to increase the already very high sturdiness of the project in terms of the transfer of heat into the atmosphere, as the ultimate heat sink. Measures leading to improved sturdiness of the units for the combination of the SBO and loss of the UHS are the same as the measures identified for an SBO and for the loss of UHS.

The aim of the proposed measures is to strengthen the defence-in-depth during initiating events beyond the design basis (earthquakes, floods, extreme conditions, human factor etc.), which could result in loss of the capacity to carry out safety functions in the case of an SBO combined with the loss of the UHS.

The opportunities to improve defence-in-depth in case of events that could result in the loss of the ability to carry out safety functions can be found in the following table. The table also contains areas that require the execution of additional analyses, because they were not available at the time of this assessment.

Tab. 34: The opportunities to improve defence-in-depth against loss of the primary ultimate heat sink, combined with a station blackout

Improvement opportunity	Corrective measure	Time period (Short-term / Medium-term II)	Note
Technical means	Alternative water supply for SG/SFSP/I.C (in case of I.C leakage)	II	
Technical means	Alternative source for recharging accumulator batteries and supplying selected appliances	I	
Technical means	Alternative supply of diesel fuel from a tank truck for long-term operation of the DG	I	
Analyses	Analysis of heat transfer from the I&C after losing the ESW	I	
Technical means	Connecting isolation valves of the containment ventilation system to the accumulator batteries	II	
Rules	Using the safety DG of the other unit in case of an SBO	I	
Analysis	Analysis of the discharging period of the accumulator batteries in case of a controlled reduction of the load, details on procedures	I	
Rules	Procedure for the isolation of the containment when in shutdown	I	SOER 2010-1
Analyses	Transfer of heat from the SFSP without an additional water supply	I	
Rules	Procedure for restoring the power supply after an SBO in all units	I	
Staff	Securing a sufficient number of staff in case of a long-term SBO	I	
Analyses	Analyses of the possibility of shift staff in case of an SBO in both units	I	
Rules	EDMG manuals for using alternative means	II	
Communication	Alternative sources and means of communication after a seismic event	I	
Rules	Executing procedures for the operation of units in case of a long-term power supply from emergency sources	I	

III.5.4 Spent fuel pool storage pools

With respect to the design solution of the VVER1000 units, with the SFSP located in the containments immediately next to the reactor, the power supply systems and transfer of heat into the ultimate heat sink from the spent fuel stored in the SFSP employ similar means of heat transfer to those used for transferring heat from the core. Therefore, the assessment of the loss of power supply and loss of UHS for the transfer of heat from the SFSP can be found in the corresponding parts of Chapter III.5.2

Although a complete failure of the heat transfer from spent fuel stored in the SFSP would require the unlikely combination of multiple failures at the level of defence-in-depth protection, the seriousness of the consequences of such a situation require that further measures be proposed to increase the already high sturdiness of the project in terms of heat transfer from the SFSP into the ultimate heat sink, regardless of whether due to an SBO or loss of the UHS.

The goal of the proposed measures is to improve the defence-in-depth protection in case of initiating events beyond the scope of the current design basis (earthquake, floods, extreme conditions, human factor, etc.), which can result in loss of the ability to carry out safety functions.

The opportunities to improve defence-in-depth in case of events that could result in the loss of the ability to carry out safety functions can be found in the table 34. The table also contains areas that require the execution of additional analyses, because they were not available at the time of this assessment.

III.6 Severe accident management

Correct understanding of the following text requires familiarity with the contents of Chapter III.1.1, which describes the technological systems designed to carry out main and auxiliary safety functions within the Temelín NPP.

III.6.1 Organization and measures of the licensee to manage accidents

III.6.1.1 Organization of the licensee to manage accidents

The existing system for coping with severe accidents includes a set of measures – personnel, administrative and technical. In the personnel area it is: Creating emergency response organization and carrying out specific activities. In the administrative area it is: Processing and implementing the necessary procedures, manuals and instructions. And in the technical area it is: Ensuring functionality in the scope necessary for the implementation of strategies and creation of emergency support centres, in which the staff controls and carries out actions. In the first (preventive) phase of an event, actions are taken by the current shift staff. In case the event grows beyond the capacity of this staff, the second phase (mitigating consequences) starts, and emergency response organization is activated. In this case, the responsibility for coordinating activities is assumed by the emergency headquarters assisted by the technical support centre.

Staffing and shift management in normal operation

The staff of the continuous shift operation of both units of the Temelín NPP (shift staff) is divided into shifts. The number of persons in each shift and their qualification ensures that all operating states are managed under all normal,abnormal and emergency conditions. Shifts are changed regularly according to the shift schedule to leave sufficient time for rest and maintaining the required qualification (training, etc.).

The shift staff carries out activities according to the operating documentation (procedures, instructions, programs, etc.), covering ordinary and extraordinary operation, as well as emergency conditions (including all design basis and, in part, also beyond design basis events until fuel degradation). During all these states of operation, the staff controls and carries out activities with the possible support of the technical personnel in the NPP. In case of emergency conditions involving fuel degradation, the responsibility for managing activities is shifted to the personnel in the TSC (technical support centre) and EH (emergency headquarters), and the shift personnel continues carrying out the instructions of the TSC and ET.

Persons in the following positions are responsible for managing the corresponding units of the NPP in case of an extraordinary event:

- Head of the reactor unit (reactor unit supervisor)
- Head of the unit control room (control room supervisor)
- Primary circuit operator
- Secondary circuit operator.

The basic workplace of these personnel is the unit control room. In case it is not usable or if control over the unit systems has been lost, the necessary activity is carried out in the emergency control centre.

During the origination of extraordinary event (EE), the SE ensures the immediate notification of management of NPP Temelín and ČEZ, immediately reports the event to SÚJB, Regional Office, Regional Directorate of FB, those municipalities with an expanded scope of authority, to Technological Dispatcher ČEZ and to the metrological station. For handing over information, the form “Primary notification, or consequent reports on the origination of an extraordinary situation” is used. Electronic mail is used for sending forms or faxes. If impossible to establish a direct connection with SÚJB then an alternative method is used.

To enable the planning of protection of inhabitants in the surroundings of NPP in the case of a radiation accident and for the requirements for preparation of an external emergency plan, on the basis of the decision of the SÚJB, a zone for emergency planning of the NPP for NPP Dukovany is declared with a radius of 20 km. For ensuring measures for the preparation and actual evacuation of inhabitants, this decision states the internal part of EPZ stated by a ring with the radius of 10 km including the municipalities on its border.

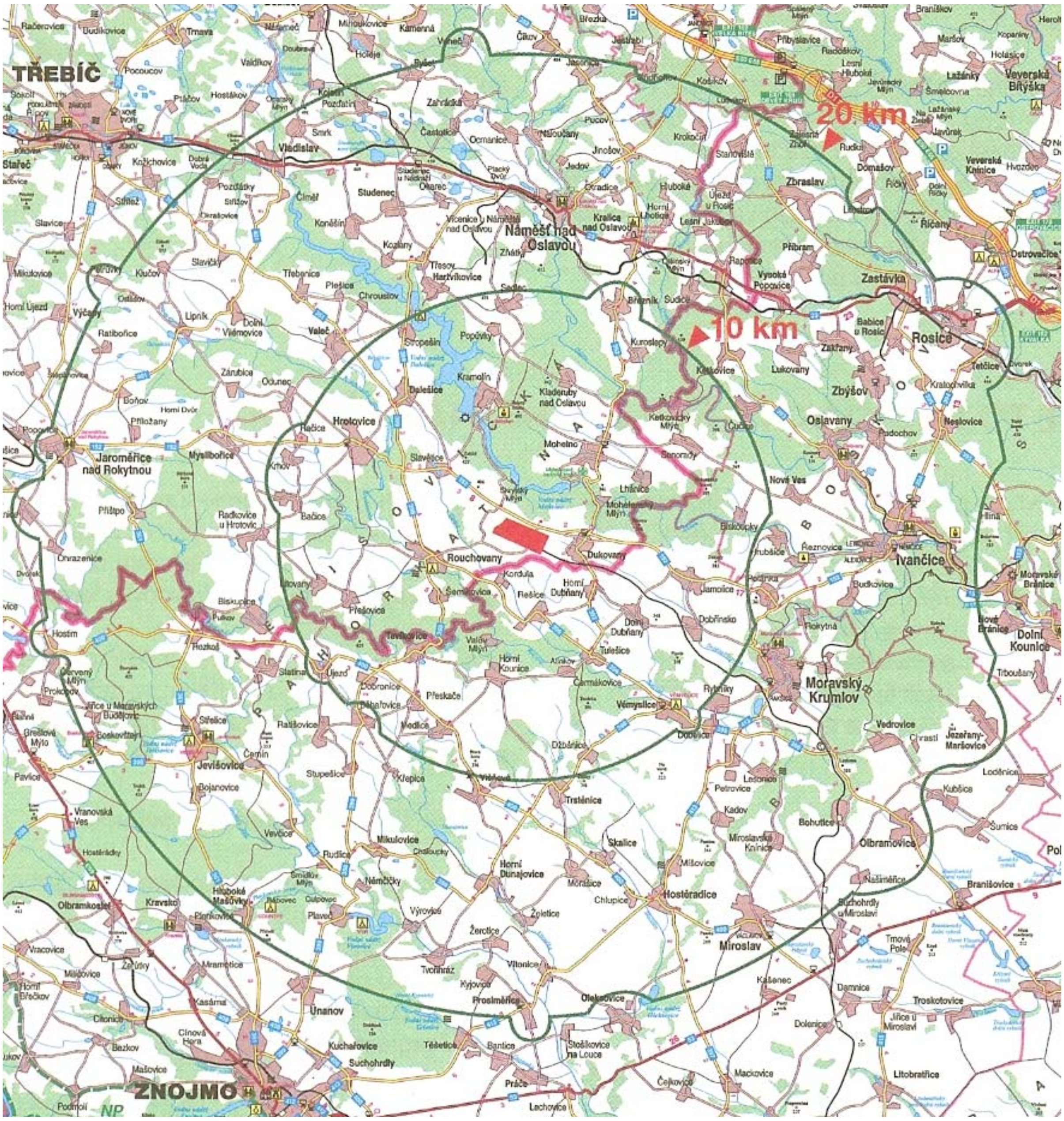


Fig. 41: Emergency planning zone(EPZ)

Measures taken to enable optimum intervention by personnel

During a threat to safety on the unit or in the site or during the origination of a situation that cannot be managed by labor forces in the shift, the shift engineer announces three levels of extraordinary events.

EE 1 level (Alert),

EE 2 level (Site emergency),

EE 3 level (General emergency).

For managing extraordinary situations, an emergency response team is created where the internal part (IOER) consists of shift personnel and the on-call section (SOER) consisting of technical experts of the NPP who should be ready for action (in 4 crews). The SOER members are on-call so that within 20 minutes during working hours and within 1 hour during off-working hours from the announcement of an extraordinary emergency event, the respective experts are in attendance at NPP Temelín to ECC. Means for activation of SOER personnel are backup.

The procedure for evaluation of the seriousness of originated extraordinary events in individual power plants is stated in the respective action instructions. The seriousness of the originated reported events is evaluated by SE by comparing the type of reported event with the set of the previously defined intervention levels. The EE can also be classified by the commander of the emergency board. The intervention levels represent the set of previously stated, locally specified, initiation conditions when the status of the nuclear power plant is classified into the respective classification level and type. The intervention levels are prepared for all operating regimes of the nuclear power plant. The initiative condition may exceed any of the stated parameters, or the occurrence of discrete internal and external events that may threaten the nuclear safety and radiation protection of the nuclear power plant.

Types of extraordinary events

The timely identification of the type of originated event and evaluation of the seriousness in terms of nuclear safety enables selection of an appropriate response. Extraordinary events are divided into three basic types:

- Events from technological reasons
- Radiation events
- Events from other risks.

This classification of intervention levels enables the shift engineer to easily identify the seriousness of the originated extraordinary event in relation to ensuring nuclear safety and radiation protection.

The declaration of EE is in the full competence of the SE. In the case of declaration of EE 1 stage, only the technical part of SOER is activated (technical supporting group - TSG); in the case of the announcement of EE of the 2 and 3 stage, the remaining part is activated – Emergency response board NPP Temelín (ERB).

SE is liable for all activities related to the resolution of EE from the time that the commander of ERB accepts responsibility for the EE solution. SE continues to be responsible for the management of technology in the undamaged units. The liquidation is managed by SE on the affected unit. SE is responsible for the fulfillment of the commands from the ERB commander in the area of the management and coordination of the activity of the work shift.

Up to the time that the commander of ERB takes responsibility, SE ensures:

- the provision of the stated information
- coordination of assistance of FB, medical readiness for action, guarding G4S
- in the case of EE level 3, issues warning to employees in the NPP Temelín site and EPZ inhabitants
- manages the safe removal of personnel, if necessary.

The workplace of TSG and ERB is the Emergency Control Centre ECC), which is located in the NPP Temelín site. In the case of the announcement of EE of the 2nd and the 3rd level, the following centres are activated:

- logistics support centre in České Budejovice (concentration, boarding and accommodation of necessary experts for the solution to the emergency situation),
- emergency information centre in Česke Budejovice (ensuring contact with journalists and keeping the public informed)

- external emergency support centre in Ceske Budejovice (ensuring radiation monitoring in EPZ). All these centres are established by the emergency board.

The organizational method of managing of extraordinary events is stated in the internal emergency plan approved by SÚJB

For the solution of technological accidents (up to damage of fuel), strategies have been prepared that are contained in the emergency operating procedures (EOPs). To mitigate the consequences of accidents after damage to fuel (major accident), strategies have been prepared and are contained in the manuals for managing major accidents (SAMG). In EOPs the main priority is the restoration of the heat removal from RC and the prevention of damage to the 1st barrier against the release of fission products (covering the fuel), whereas in SAMG the main priority is the prevention of damage to the 3rd barrier against the release of fission products (containment), which at this time is the last integrated barrier.

Internal organization of emergency response

The internal emergency response team consists exclusively of shift personnel, i.e. employees who work on the normal operation of the NPP. Personnel from the permanent shift ensure, according to the instructions of the shift engineer, all activities related to the suppression of the effects of the originated extraordinary event up to the time of the activation of the on-call team of employees who are ready for action within the organization of the emergency response.

The shift engineer, in the case of the origination of EE, is responsible for the announcement of EE and the management of activities in accordance with EE up to the time when responsibility is handed over to the activated ERB commander. His activity during the origination of EE is managed according to instructions for shift engineer which lists all responsibilities and authority, of which the most important are: evaluation of the seriousness of EE – classification, ensuring notification and warning of personnel of the NPP and warning in EPZ, notification of the management of the NPP and the respective bodies and organizations about the origination of EE, decision on the activation of SOER, decision on protection measures for the NPP personnel. The responsibility for technology continues to be in the competence of SE.

The shift personnel (with the exception of the personnel for the shift on MCR) in the case of the announcement of an extraordinary event depending on the level of seriousness, either executes the activity according to the respective intervention instructions and the instructions of the management personnel of the shift, or are concentrated in the case of the declaration of protective measures in the operating support centre in the shelter under the operating building, from where on the basis of SE or ERB they make the required interventions in the technology or create operative support for the FB unit during the recovery and rescue work.

For the requirement to ensure the implementation of protective measures for hiding and evacuation, teams are established which ensure the activation and the consequent operation of shelters in the NPP site. The basic obligations of members of the hiding teams in the shelters are: management of the regime in the hiding place, record keeping of hiding places, order service, operation of the air ventilation system, dosimetric measurement of persons servicing DGs.

Emergency organization of emergency response

The emergency response organization consists of personnel of the emergency supporting services ensuring week-to-week continuous preparation for action.

- Emergency board

  The emergency board is the main management workplace of the emergency response organization of the NPP. After activation it ensures the announcement of protective measures for employees and other persons located in the site of the NPP at the time of the origination of a extraordinary event, management of activity of all employees and other persons participating in the action during the suppression of the development and resolution of the consequences of an extraordinary event in the nuclear power plant and ensures communication with external elements of emergency preparation. The emergency board ensures the delivery of necessary material, special means, alteration of personnel and material ensured through the logistic support centre.

- Technical support centre
  The technical support centre is professionally staffed to provide qualified technical support for the personnel of the control room for the affected unit during the resolution of extraordinary events. At the same time, staff of TSC ensures the immediate evaluation of the security status of the nuclear power plant with an emphasis on nuclear safety and radiation protection, manages the activity of the operatively stated action groups during the resolution of the consequences of extraordinary events and is able to prepare source materials and recommendations for making decisions and controlling the activity of the emergency board. The head of TSC may request through SE or the commander of the emergency board the strengthening of TSC by further experts.

- External emergency support centre
  External emergency support centre ensures activities related to radiation monitoring and evaluation of the radiation situation in the emergency planning zone and on the basis of the results of radiation monitoring and the prognosis of the further development of the radiation situation.

- Emergency information centre
  The personnel emergency information centres ensure, in the case of origination of an extraordinary event, the distribution of information to mass media and answering questions from the public. Its activity is to inform the general public and state administration bodies and self administration not involved in the system of external emergency, about the status of the nuclear power plant. It is responsible for the preparation of press releases for mass communication media. The emergency information centre is located on the premises Health and Social Faculty of the South Bohemian University in České Budějovice.

- Logistics support centre
  The personnel in the logistics support centre ensures the necessary means of technical material and highly-qualified staff according to the requirements and demands of the emergency board, technical support centre and external emergency support centre. The logistic support centre represents the external ERO support. The logistic support centre is located on the premises Health .and Social fakulty of the South Bohemian University in České Budejovice

Use of off-site technical support for accident management

Ensuring of external support and use of further capacities, resources and means is managed in ERB by the logistics employees, in cooperation with the logistic support centre.

For assistance with transport or heavy technology, the integrated rescue system of the Czech Republic will be used with the control centre at FB for the South Bohemian region, which has the authority within IRS to call on further elements and organizations to help when managing the consequences of an extraordinary event. Within the whole ČEZ group, for the affected site, help is stated through the crisis board of ČEZ for the affected area. Within this body, availability of external experts is ensured (suppliers, experts from research institutes ÚJV Řež, EGP, VÚJE with knowledge of the respective issues, foreign help from other NPP of the VVER type in the localities of the power plants at Bohunice, Mochovce, etc.). The most effective help is expected to be from Dukovany NPP.

The external preparedness of the NPP is ensured with the participation of a series of bodies and organizations at both a national and local level.

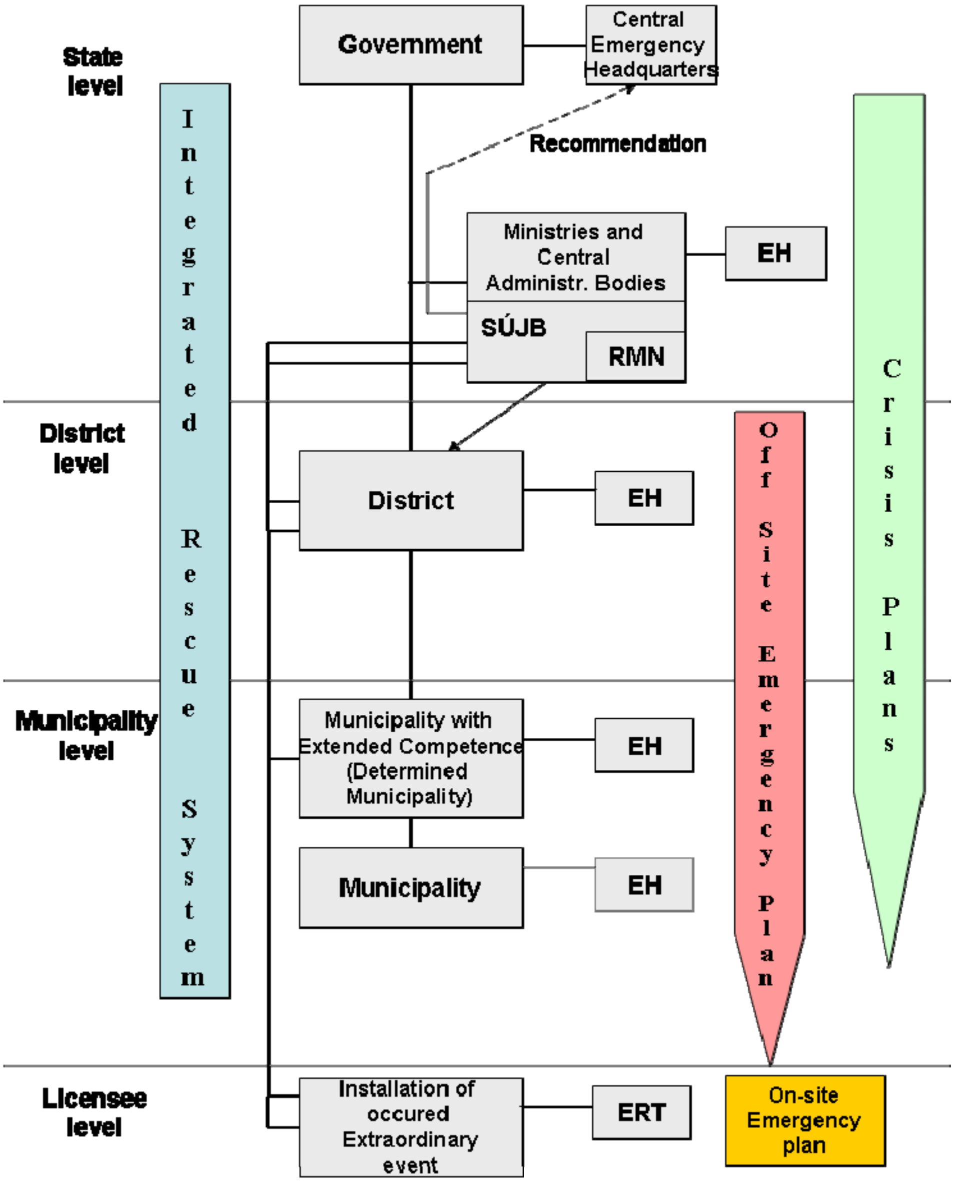


Fig. 42: Ensuring the external emergency preparedness of the NPP

During the occurrence of EE and the consequent resolution of the originated EE, the nuclear power plant communicates with external bodies and organizations at a national and local level.

- SÚJB – Crisis board
  The SÚJB crisis board ensures, through radiation monitoring of the network of the Czech Republic, an independent evaluation of radiation effects of the originated extraordinary radiation event. On the basis of the results of the monitoring of individual elements of the network of the Czech Republic, it provides source materials for decisions by the crisis board for the region on measures for the protection of inhabitants.
- Regional government

The regional government ensures coordination of the external emergency preparedness of all villages with expanded authority where the territory intervenes into EPZ. The supervisor of the respective region, in cooperation with the mayors of the affected villages with expanded authority, manages all activities related to ensuring external emergency preparedness in the whole emergency planning zone and decides on the announcement and implementation of measures for the protection of inhabitants. The crisis board for the region serves as an advisory body for the supervisor. The declaration of urgent protective measures is made on the basis of the recommendations of the SÚJB crisis board prepared from the results of radiation monitoring and other resource materials provided by individual elements of the radiation monitoring network.

The operator provides, in the case of a radiation accident in the nuclear power plant, for the crisis board of the region and through the emergency board, the necessary assistance, data and information required to evaluate the seriousness of the originated situation. To ensure assistance, the nuclear power plant will send a representative to the crisis board of the region.

- Villages with expanded scope of authority
  The mayors of affected villages with expanded authority will decide on the calling up of the crisis board for their village and manage the announcement of protective measures in the concerned territory of the village with expanded authority. During the management of these activities, they use the external emergency plan. Protective measures are announced, after prior discussion with the crisis board of the region, which ensure the coordination of messages and information passed between individual villages with expanded authority, SÚJB and the nuclear power plant. This procedure serves to ensure the line of announced protective measures in the territory located under the management of individual villages with expanded authority.

- Czech Hydrometeorological Institute
  The Czech Hydrometeorological Institute evaluates for nuclear power plants the actual hydrometeorological situation and prepares a prognosis for further development. The outputs of basic hydrometeorological data necessary for the evaluation of the potential or actual spread of radioactive releases in the surroundings of the nuclear power plant are handed over to the respective information networks of the nuclear power plant.

- IRS – Integrated rescue system
  The integrated rescue system (hereinafter referred to as IRS) exists for the purpose of the coordinated management and the solution of extraordinary situations on the basis of detailed specification that it concerns industrial discrepancy, floods, earthquakes or other natural disasters. In terms of legislation, this issue is resolved in the laws on integrated rescue systems and crisis management. Within IRS, a central IRS alarm plan is prepared which will be used when necessary, due to the extraordinary event of the crisis situation or a safety event, and when the legally stated conditions are fulfilled for the central coordination of rescue and liquidation work, or if the regional supervisor, the mayor of the village with expanded authority, director of FB of the region or the commander for action required through the ITS operating and information centre of the region for help and labor forces which IRS units do not have at a regional level the means for performing rescue and liquidation work during the extraordinary event to be resolved independently in the respective region.

  Labor forces and the means for the central coordination of rescue and liquidation work are called and implemented by the Ministry of the Interior – General Directorate of the Fire Rescue Brigade of the Czech Republic (hereinafter referred to as the "General Directorate") through its operating and information centre.

- Fire rescue brigade

The fire rescue brigade, on the basis of instruction from the nuclear power plant, is responsible for warning inhabitants in the emergency planning zone using sirens controlled through the national integrated warning system and also ensuring broadcasting of the respective radio and television broadcasts at ČT and ČRo. On behalf of ČEZ, a. s., FB also notifies the concerned villages with expanded authority through the regional operating and information FB centres (in accordance with Regulation No. 318/2002 Coll., as amended).

- Police, security service and army
  Within IRS, among others, 6 helicopters are reserved for rescue work (Army o fthe CR and the Police of the ČR) with the possibility to transport persons and loads where 4 crews are in the ready for operation status with the possible activation within 10 minutes during the day and 20 minutes during the night.
- Rescue medical service (Traumatological plan)

Procedures, training and exercises

The concept of managing technological accidents at NPP Temelín is based on symptomatic aproach At present, the following strategies are prepared for NPP Temelín for the solution to the above-project and severe accidents.

- Symptomatically oriented emergency operating procedures for power modes (EOPs).
- Symptomatically oriented emergency operating procedures for shutdown modes, including cases of threats to the heat removal from the spent fuel in SFSP (SDEOPs).
- Manuals for decision making by TSC
- Severe Accident Management Guidelines (SAMG).

All the mentioned regulations and manuals were developed and are updated in cooperation with the Westinghouse company.

The procedure for the development of an emergency situation is, in addition to the activities of the emergency response procedure, in accordance with the internal emergency plan (announcement of the level of an extraordinary event).

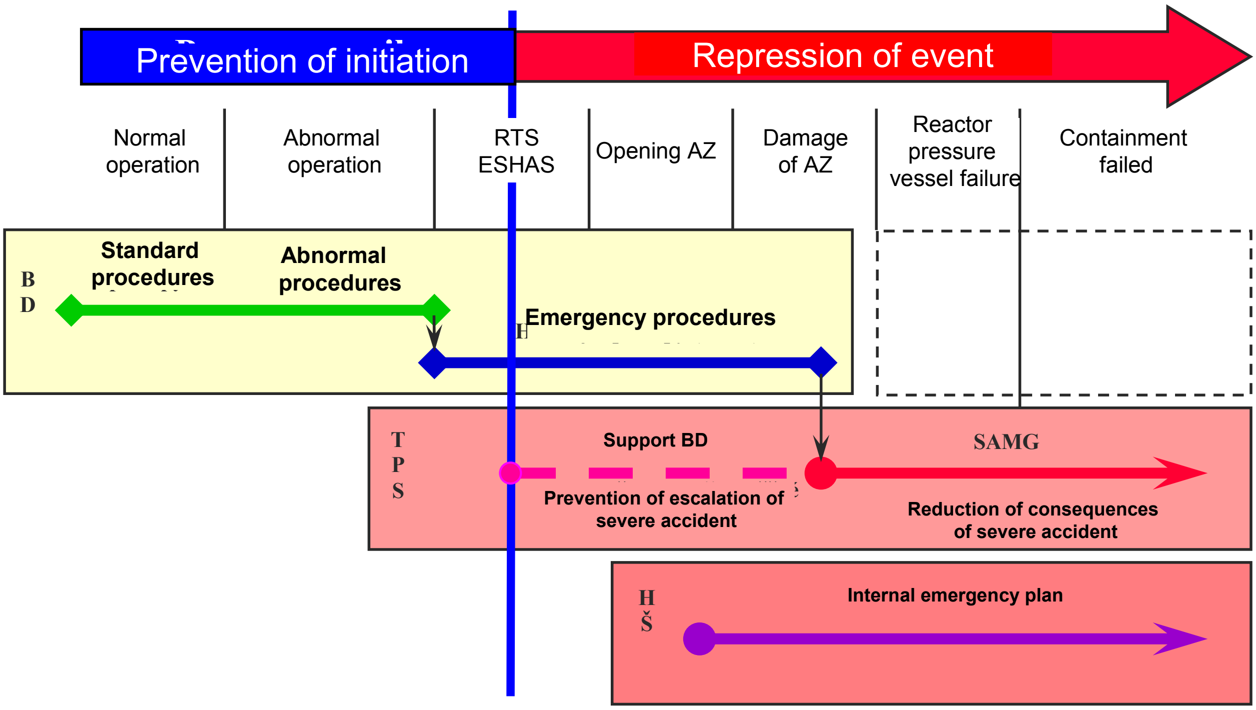


Fig. 43: Relation between the status of the unit, operating documentation and EE

The activities of the operative personnel at each level are managed by the operating procedures adapted to each operating status. The procedures represent the network which states the activities of the operating personnel in each operating status of the unit.

For emergency conditions in the preventive phase, strategies are prepared that are included in the emergency operating procedures (EOPs). For managing severe accidents, strategies are prepared which are included in the Severe Accident Management Guidelines (SAMG). The basic condition for executing the activity according to emergency procedures is that the status of RC enables cooling, i.e. RC is a geometric configuration that can be cooled. In the case of irrevocable damage, emergency regulations need not provide the optimal manual for the solution of the emergency situation and it is necessary to proceed according to SAMG. At this time, the main priorities are changed. In EOPs the main priority is the restoration of the heat removal from RC and the prevention of damage to the 1st barrier against the release of fission products (covering the fuel), whereas in SAMG the main priority is the prevention of damage to the 3rd barrier against the release of fission products (containment), which at this time is the last integrated barrier.
The objective of interventions described within EOPs, which the operating personnel will use for the solution of the project and above-project emergency events, is to ensure sufficient cooling of RC and prevent irrevocable damage to RC and also minimize the consequences of any release of radioactive substances outside the NPP. The philosophy for this procedure includes the continuous evaluation of critical safety functions monitoring the statuses of physical barriers against loss of activity. This evaluation ensures the timely identification of the worsening of the safety status of the unit and ensures timely correction in the case of ascertaining a negative trend in the development of the event.

The set of symptomatic oriented emergency operating procedures provides a systematic tool (independent to the course of the emergency regime) for operative personnel to report emergency situations by means of a set of previously stated and structured emergency procedures. The combination of event and function oriented strategies provides operating personnel with guidelines for putting the unit into safety and end status while ensuring continuous diagnostics of the status of the unit and the restoration of the safety status independently of the course of the stated emergency event.

The emergency procedures also contain systematic means for the evaluation of the safe status of the unit through the evaluation of the statuses of critical safety functions. The critical safety functions closely relate to the physical barriers which prevent the release of radioactivity into the surrounding environment. If the integrity of the matrix is ensured, the coverage of fuel and the interface of the primary circuit and the containment is ensured, then the power plant will not endanger the health and safety of the inhabitants. However, in the case of breaching of one or more barriers, then the risk of endangering inhabitants will increase. In the case of the loss of the integrity of all barriers, there would be a direct threat to inhabitants and it would be necessary to take extraordinary external emergency actions. For these reason the objective of the operation of the nuclear power plant (pursuant to nuclear safety) is to ensure to the maximum level and under any conditions or events, the integrity of physical barriers preventing the radioactive releases.. .

For the preventive phase of managing the emergency situation when the operating personnel does not proceed according to EOPs, the TSC personnel has manuals available ("Manuals for TSC) containing source materials for making decisions during the support of operating personnel executing activity according to emergency procedures. The emergency procedures include many steps that explicitly require instructions for further activity from TSC personnel. At the same time, experience from training, simulation on a full-scope simulator, etc., shows that support from TSC personnel is required in a series of other situations that do not explicitly require it. In all such cases, the decision depends on the actual development of the emergency situation and the specific status of the systems and equipment in the units. These manuals were created for personnel and for further NPP personnel who, in addition to TSC personnel, are entitled to provide support for decision making.

- Manuals for TSC are used by TSC personnel; when MU1 has been announced, TSC has been called and TSC personnel are able to provide support.
- Manuals are used by safety supervisor, SE or reactor unit supervisor, if support for a decision is requested before then, TSC becomes functional.

TSC personnel provide support by evaluating the actual status according to the manuals and by handing over recommendations during emergency procedures.

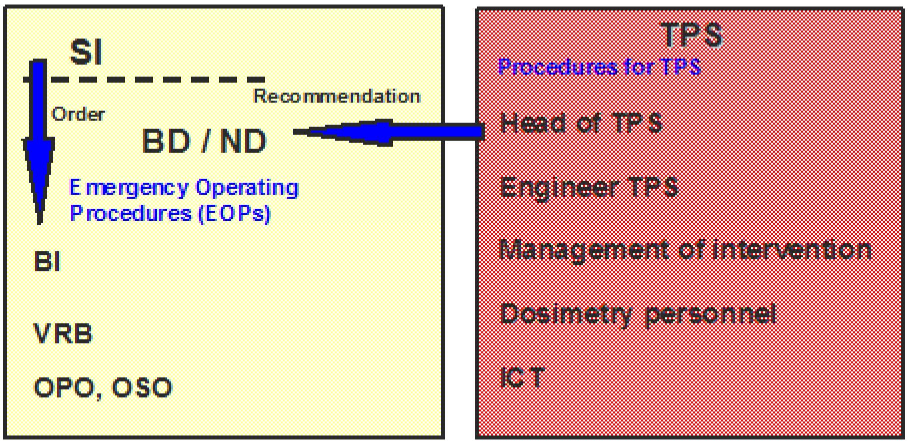


Fig. 44: A diagram of communication between TSG and the operating personnel when using the Manuals for TSG

The validity of the emergency operating procedures are restricted in relation to the precondition for keeping such geometry of RC that enables heat to be collected from the fuel. The strategy for emergency operating procedures provides manuals for solution of the statuses of the unit up to the commencement of the degradation of structures inside the reactor. Keeping the integrity of internal reactor parts and preconditions for keeping the compact character of the fuel in RC is the basic condition for ensuring the heat removal from RC to prevent the melting of fuel. In other words, as long as there is no such development of a scenario that would lead to severe damage to the fuel in RC, then it is possible to proceed according to the set of symptomatic oriented emergency procedures.

In the case of the development of an event in the area of the severe accident, a further procedure is selected to at least ensure the remaining barriers against the release of radioactivity. The loss of integrity and geometry of the fuel in RC means a serious threat to the ability to remove heat from RC. Under these conditions it is not possible to proceed according to EOPs. SAMG for achieving stabilized status are prepared for this phase of the accident.

The transfer to SAMG takes place in the case that irrevocable damage to RC is ascertained. In this case activities according to EOPs are terminated and the transfer to SAMG takes place.

There are three possible transfers from EOPs into SAMG:

FR-C.1 Loss of cooling RC

FR-S.1 No shutdown of reactor (ATWS)

ECA-0.0 Loss of electricity supply – Blackout.

These three transfers for emergency statuses into SAMG are sufficient and cover all possible severe accident scenarios.

To decrease the consequences of severe accidents, the following objectives must be fulfilled:

- **Primary objectives of SAMG**
  - Restoration of the heat removal from RC or from melt = restoring the source of the development of heat to a stable status that can be controlled.
  - Keeping the integrity of the containment as the last barrier against the release of Ra substances into the surroundings = ensuring the status of containment that can be controlled
  - Terminate the release of Ra substances into the surroundings
- **Secondary objectives of SAMG**
  - Minimizing the release of Ra substances while fulfilling the primary objectives
  - Ensuring the maximum operating capability of equipment during the fulfillment of the primary objectives.

For managing severe accidents, the symptomatic oriented approach is consistently applied. The basic principle of this approach is that the respective strategy of the solution is selected on the basis of the actual development of the accident which is identified on the basis of unique symptoms (features). If during the solution of the accident there is a change in the symptoms and the chosen strategy cannot be used, then the structure of procedures and manuals enables changing the original strategy and continuing with activities stated by another procedure or manual which better corresponds to the originated conditions. The continuous diagnosis of the status of the unit during the accident enables correct response to changing conditions in the development of the accident and interventions are always an optimal response to the status of the unit and also take into consideration the external event and the risks.

In the case of activity according to emergency procedures being terminated and the transfer into SAMG, activities according to the manuals for TSG are terminated and for further management of the activity, only SAMG are used.

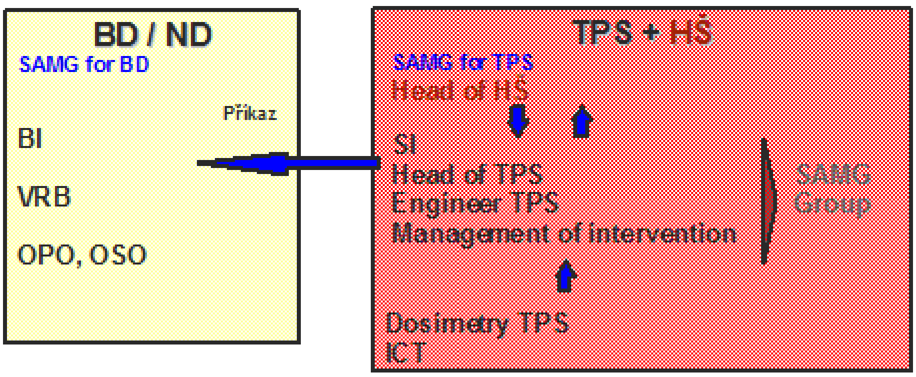


Fig. 45: Diagram of communication between TSC and operating personnel during the use of SAMG

To date there are no manuals for NPP Temelín for managing severe accidents for shutdown statuses (SAMG for shutdown statuses) designated for statuses with above-project emergency situations due to damaged fuel in RC, which develop into a severe accident during the shutdown of the reactor (open reactor) or for the above-project accident situation due to damage to fuel in SFSP

For maintenance EOPs, SDEOPs and SAMG maintenance there are regular updates that include knowledge from their use on the simulator or during emergency drills. External knowledge (within the “users group” and long-term cooperation with Westinghouse) is included in this documentation in the form of the “Maintenance program”.

For the relevant workplaces for the shift and supporting personnel and for execution of activities when managing accidents, personnel and qualification conditions are stated and these requirements are also checked through the set of qualification preconditions. For each workplace there are requirements for education, specific knowledge (basic preparation, periodical preparation, simulator training, training of selected personnel from the area of the above-project and severe accidents) and professional development training. A system of requirements for qualification is introduced for selection of SOER personnel, and further criteria taking into consideration their knowledge and professional orientation are considered. For improvement of selection and training of personnel involved in SAM it is necessary to adopt further measures.

The preparedness of the shift and technical personnel for managing technological accidents (including the transfer from EOPs to SAMG) is regularly checked during training on the full-range simulator with the participation of TSC personnel (2 per year) and during emergency training.

Emergency exercises take place a minimum of 4 times a year so that each SOER shift passes the exercise at least once a year. The exercises also include preparation for versions of preventive actions in severe conditions. The respective procedures were prepared for the activities of the intervention groups in severe conditions and for their protection. The real drill in the use of SAMG when managing severe accidents on NPP Temelín took place after putting SAMG into use (under the supervision of Westinghouse).

As the full-range simulator is not designed for a simulated course of severe accidents, a new simulating tool is being developed with UJV Řež enabling the display of courses of the parameters and their behavior in time and space. It is one of the measures resulting from the Periodical Safety Review. This simulation tool will be used during the creation, update and maintenance of SAMG and, at the same time, will also be used for training the personnel of Temelin NPP (mainly TSC) in the use of SAMG procedures for managing severe accidents. The tool is based on an animated display of the course of a severe accident in the reactor, the primary circuit and the containment. The display will be interactive: according to selection, it is possible to change the speed of the display, repeat selected sections of the accident and select the set of additional animated graphs of the characteristics of the accident.

Dependence on the functions of other reactors on the same site

Both units of the Temelín NPP are technologically independent and constructionally separate. Common facilities of both units include: Raw water supply from the Vltava River and the CBSS for transferring heat from the reactor core, the SFSP and safety systems providing the transfer of heat into the ultimate heat sink – the atmosphere.

In case the raw water supply for the NPP is lost, each of the three CBSS is able to transfer all heat from both units for 12.5 days without an additional supply of water. An analysis of the utilization of mobile fire equipment in the location showed that the water for the running CBSS can also be supplied from other CBSS using mobile pumps.

Besides the CBSS (passive, seismically resilient objects), all other technological systems transferring heat are independent and constructionally separate for both units.

Thanks to the independence of the power supply from external and internal sources (including emergency) from each unit, it is possible to use the power sources of one unit for the other unit, which is undergoing an SBO.

Another common facility that could be relevant for coping with severe accidents is the reserve of boric acid solution, stored for both units in the auxiliary building (BAPP), which contains additional 1600 m^3 available for both units of the NPP (volume comparable to the volume of boric acid solution kept in the containment reservoir).

The containment of each unit contains an SFSP. This layout is expedient because it prevents fission products from escaping in case the fuel stored in the SFSP is damaged. The disadvantage is the difficult access to the SFSP for emergency resupply (fire equipment etc.). Also, the SFSP could be affected by an accident within a system located in the containment or vice versa.

III.6.1.2 Possibility to use existing equipment

Provisions to use mobile devices (availability of such devices, time to bring them on site and put them in operation)

Symptomatic procedures and guidlines (EOP and SAMG) were executed to ensure safety functions for design basis and beyond design basis scenarios. However, neither the EOP nor the SAMG includes the possibility of using mobile or non-technological means from external sources (besides the means of the LFRU – local fire rescue unit).

One brigade of the fire rescue unit with approximately 16 firemen per shift (permanent for 24 hours) is available in the site of the Temelín NPP. The unit has adequate fire equipment and is trained to cope with fire anywhere within the location.

The pumps of the LFRU are among the main mobile non-technological means usable for the transport and pumping of media. These pumps have also been adjusted for pumping water in case of floods. For pumping and transporting water, the fire rescue unit has 4 tank trucks, 1 combined fire vehicle and 3 trailer pumps with a nominal output of 280 l/s.

However, until now this equipment has not been considered for use in mitigating the consequences of technological accidents. Therefore, the capacity of this means has not been verified and connections that would allow connecting this equipment with the systems for basic safety functions have not been prepared.

The fire equipment and personnel are located in the building's Fire Station, which protects them from the direct effects of natural events. However, the ability of the fire unit to exit the building and reach the accident location could be limited. In case of a seismic event, it is expected that the fire brigade would immediately, at the first signs of an earthquake, move out to open space, where the equipment and personnel are protected from falling buildings and structures.

Provisions for and management of supplies (fuel for diesel generators, water, etc.)

Diesel fuel management (4 tanks, each 1,000 m^3) is available in the location, ready to supply long-term operation of the DG. However, it can also be used in case other mobile diesel generators are involved. All diesel generators available in the location have internal tanks with diesel fuel for autonomous operation (without adding more diesel fuel) under maximum load:

- Safety DG for at least 48 hours (realistically longer),
- Common DG (supply of appliances in both units) for about 12 hours.

In case long-term operation is needed, more diesel fuel can be brought by a piping leading along technological bridges from the diesel fuel management. However, the pumps of diesel fuel management are not supplied from a secure power supply; therefore, a further supply of diesel fuel would be provided by tank trucks.

All auxiliary systems of the DG's engine and generator (fuel intake, lubricating oil, internal cooling circuit, air, starting air) are autonomous and, while the DG is in operation, independent of the external power supply. Systems that could be negatively influenced by long-term operation of the DG (e.g. clogging of the filter for the oil system) have redundant subsystems, so that one can be shut down while the DG is in operation to carry out maintenance and thus prevent failure of the DG due to a loss of the auxiliary systems. The quality of the diesel fuel is checked monthly and kept in accordance with the corresponding requirements.

To fill demi water in the SG, there are 3 x 500 m^3 reserve tanks of the emergency supply system for the SG available for each unit, as well as 2 x 770 m^3 tanks common for both units. This amount of water is sufficient for cooling the units to a cold state (according to the project, one system of emergency supply for an SG is sufficient for cooling the unit to a cold state) or maintaining the unit in a hot state for about 72 hours.

Even in the worst-case scenario, each CBSS is capable of transferring, for a long period of time, all heat from a unit with a shut down reactor without a resupply, so that the temperature of the ESW does not significantly exceed the maximum design value. The worst case scenario is a LOCA on one unit and shutdown procedure running on the other unit, i.e. the heat transferred via the ESW is peaking. Because there are three redundant ESW systems, it has been shown that heat can be transferred into the ultimate heat sink for at least 30 days, assuming that the safety divisions will be used one by one or that the water from an inoperable CBSS will be transferred by mobile means into the CBSS of a functional ESW system.

Management of radioactive release, provisions for limiting them

The goal of all strategies for managing accidents (in the EOP and SAMG) is to prevent radioactive release into the environment and thus protect the health and safety of people. However, should there be radioactive release, during an accident, all activities will be directed toward terminating, or at least limiting the release,

The Temelín NPP uses a program called RTARC for creating a prognosis of the consequences of a potential radioactive release as well as for assessing the current situation. RTARC uses most recent weather data, weather forecasts, data about the radioactive leak, dimensions of the main production unit (MPU), data about the terrain surrounding the NPP and data about escaping radionuclides. RTARC works with the data from pre-calculated source parameter, which can be adjusted depending on the current measurements. The output is the current radiological situation and prognosis of the situation for selected period from 500 m to 40 km around the NPP.

Limiting the radiation exposure for the population and the environment in case of an extraordinary radiation event involves protective measures:

- Immediate measures involving sheltering, iodine prophylaxis and evacuation.
- Subsequent protective measures, including resettlement, regulation of the use of food and water contaminated by radionuclides, and regulation of feed material contaminated by radionuclides.

Protective measures in case of radiation accidents are carried out whenever their benefit exceeds the costs of measures and damage inflicted. These measures must also be optimized in terms of the form, content and duration, thus providing the maximum reasonable benefit.

Depending on the level of radiation in case of an extraordinary event, protective measures will be announced in such a manner that the radiation dosage levels in the following tables are not exceeded. The table contains limits on radiation dosage for employees and other persons after announcing protective measures in case of an extraordinary event, 8 hours after the event has occurred.

These limits do not apply to persons participating in mitigation of potential radiation accident, but even for such persons the radiation doses must not exceed ten times the basic limit for employees working with sources (the basic limit is 50 mSv for the calendar year or 100 mSv for 5 consecutive calendar years), unless there is an attempt to save human lives or prevent a radiation accident with large-scale social and economic consequences. Employees who would be carrying out the action would be informed of the risks and expected amount of radiation to be received.

Tab. 35: Protective measures for the staff in the NPP

	PROTECTIVE MEASURE		
CATEGORY OF PERSONS	**Shelter**	**Iodine prophylaxis**	**Evacuation**
Other persons and employees not included in OER	5 mSv	5 mSv	5 mSv
OER staff	50 mSv	5 mSv	200 mSv
OER staff in an effort to save human lives or prevent the development of a large-scale radiation situation	According to Section 4, Subsection 7, Letter c, Act No. 18/1997 Coll.	5 mSv	According to Section 4, Subsection 7, Letter c, Act No. 18/1997 Coll.

There are 4 shelters within the NPP. One shelter contains the ECC and the other the operating support centre, where shift personnel needed for local activities would assemble. The capacity of the shelters within the NPP is 1,775 persons. Each employee in the workplace or in the shelter has his or her own means of protection to prevent external and internal contamination.

Persons are protected by means of iodine prophylaxis – taking potassium iodine pills given to all employees of the NPP as well as to the people living in the EPZ. The people living in the EPZ are also given a manual for the protection of the population, which contains instructions for a case of a radiation accident; instructions in case of a siren warning, instructions of preliminary protective measures (finding shelter, iodine prophylaxis and evacuation).

Communication and information systems (internal and external).

List of internal means of communication:

- System for warning and informing staff (sirens and internal radio)
- Telephone switchboard
- Automatic telephone machine
- Radio networks
- Paging system
- Amplifier of radio waves (emitting cables)
- Manual radio station
- Mobile radio station
- Communication system for calling the SOER and input terminal Alarm

List of external means of communication:

- System for warning and informing the population (sirens in the EPZ)
- Prepared messages for national means of communication (television, radio)

- O2 telecommunications network (mobile and fixed-line)

All communication systems are checked and verified regularly:

- Once every three months, the functionality of technical means of the communication systems and methods of activation of persons involved in controlling and carrying out response action.
- Once every six months, the functionality of technical means of communication systems and methods of warning the employees and other persons within the NPP compound.
- Once every three month, the functionality of the technical means, systems and methods of communicating extraordinary events and providing information about a radiation accident.
- Twelve times per year, the functionality of the technical means, systems and methods of warning the population in the zone of emergency planning.

Records confirming the outcomes of inspections aiming to verify the functionality of technical means, systems and methods of communication and warning are kept in an archive for three years.

There are at least two means for the activation of emergency response organizations and for starting sirens (autonomous and independent of the potential overload of mobile networks).

The emergency control centre contains an information system that provides access to all information necessary for controlling extraordinary events. In case of an extraordinary event, there is visual and acoustic contact between the TSC and both CR. Available in the CR and in the TSC is an industrial television system for monitoring key places within the NPP. The staff in the TSC has the same real-time technical and radiation data as that used by the operating control staff.

III.6.1.3 Evaluation of factors that may impede accident management and respective contingencies

Extensive destruction of the infrastructure or flooding around the installation that hinders access to the site

Access to key objects could be limited by debris from fallen seismically fragile objects on the on-site access communications, as well as from debris in the area around the access to the NPP. In such a case, it is possible to use backup entry/access to the compound.

Loss of communication facilities/systems

Backup power supply for the operation of the means of communication for on-site warning, as well as for connecting key persons (ECC, shelters, LFRU, SNSA, IRS, personnel of the CR) is provided in a matter of hours in case of a power failure or damage to the infrastructure. on the on-site sirens have backup power supply, the on-site radio has no backup power supply.

In case of a long-term SBO, the power supply for the telephone centre in the Temelín NPP and telephone centres of cooperating workplaces outside of the Temelín NPP could be lost (this does not apply to HDP ČEPS Praha and ZDP ČEPS Ostrava, which have their own DG). This poses a threat to recovering the power supply from the external network.

Restoring the power supply from sources outside the NPP (e.g. from the Lipno power plant) requires the cooperation (connection) of several external subjects (ČEZ, ČEPS, E.ON).

The fixed-line public telephone network, mobile telephone network, radio transmitters, warning systems etc. are not resistant to global floods. Possible communication via the LFRU radio transmitters.

Large-scale damage to the infrastructure could damage long-term communication between the control centres and persons carrying out the response action, as well as the communication with external control centres of state administration (REH SÚJB, regional emergency headquarters, IRS, etc.).

Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities on site

In case of a large-scale infrastructure disruption on-site and near the site, the emergency control centre in the location is to be used. If the ECC cannot be used on the site, there is a backup centre in České Budějovice (outside the Temelín NPP EPZ), which provides a limited amount of information for managing extraordinary situations. Potential inaccessibility of shelters would be approached operatively, by evacuating non-essential personnel from the location. The inaccessibility of the technical support centre could be solved by running the necessary activities from the unit or emergency control room. However, no instructions are available for these scenarios.

In case the power plant is inaccessible, the rotation of staff would be limited; personnel would sleep on the site or near the site (in shelters and ECC, possibly in the information centre).
In case of an emergency, the staff in continuous shift operation would either continue to carry out activities according to the applicable instructions or, in case protection measures were announced, would assemble in the operating support centre, depending on the seriousness of the situation. From the operating support centre (i.e. in the shelter reserved for this purpose) the staff would carry out response activities within the systems of the NPP or provide operative support for the LFRU for rescue operations.
In case the containment routes are opened, the gases released from the I.C (containing a considerable amount of fission products) would release into the hall surrounding the containment. In addition, the coolant pumped using the emergency supply systems from the reservoir in the containment contains large amount of fission products. The coolant injected into the containment could also release into the hall surrounding the containment via standard leaks in the spray system. If manipulations were to be carried out at this time, it would be necessary to protect the members of the response units, because the habitability of certain parts could be limited.

Each shelter contains systems for the protection of persons against radiation, substances of chemical and biological warfare. Thanks to their construction, these shelters provide protection against strong radiation. Technical facilities within the shelter allow their operation for at least 72 hours. In a basic setup, the shelters contain dosimetric devices for measuring surface contamination and dosage, spare emergency means of protection, spare clothing, iodine prophylaxis and means of communication. Spare emergency means of protection, spare clothing and medical material are distributed by members of the shelter team, based on justified needs and the requests of sheltered persons.

There is no heavy equipment directly on-site that would clean up debris from pivotal and access communications that could be blocked by debris from seismically fragile objects. This could complicate the access of mobile equipment to the main production units. There is a link to the possible use of heavy equipment via the IRS.

Impact on the accessibility and habitability of the main and secondary control rooms, measures to be taken to avoid or manage this situation

The rooms of the CR and ECR are located in a clear part of the hall surrounding the reactor building. This section could be hit by radiation in case of large-scale radioactive leaks from the containment. However, the CR and ECR have venting systems with filtration supplied from the SPSS; therefore, they are protected and habitable even in case of radioactive leaks.

If the CR cannot be used, the operative control staff would move to ECR, from which they can monitor the parameters of operation and control the components of the technological systems as well as from the CR.

If conditions in the CR worsen, the operating personnel would act to extend the possible time the personnel can stay in the CR (e.g. deactivation of standard venting and activation of filtered venting using aerosol and iodine filters) and if necessary, means of anti-radiation protection (protective clothing, breathing apparatuses, etc.) would be used.

If the reactor core is damaged, fission products could release into the hall surrounding the reactor, and thus the accessibility and habitability of persons in certain rooms would be limited. If it is necessary at this time to enter these premises to carry out certain tasks, the members of local response units will use the necessary protection, mainly means of protection against radiation, screening protection, protection by distance, limiting the time, etc.).

Impact on the different premises used by the crisis teams or for which access would be necessary for management of the accident

In case of an extraordinary event, all essential activities would be controlled and carried out from protected locations. The technical support centre managed by the SE and EH according to the SAMG is located in the emergency control centre. Remote activities required by the strategies would be carried out by operating personnel from the CR or ECR. On-site activities and any necessary repairs in clean parts of the reactor chamber, machine room or external objects would be carried out by response units stationed in the operating support centre (after the identification of risks, under precisely defined conditions and limitations).

The emergency control centre is a secure workplace offering permanent habitability for at least 72 hours without external support. The ECC has own sources of power supply, filtered ventilation and the possibility of being isolated from the outside. The centre contains supplies of water and frozen food. The internal environment in the ECC is monitored by dosimetric devices, and if the limit values are exceeded, an isolation mode is initiated.

Feasibility and effectiveness of accident management measures under conditions of external hazards (earthquakes, floods)

There are no deficiencies in the fulfillment of safety functions for the case of external threat to the NPP.

In order to ensure the fulfillment of safety functions, certain procedures and strategies were executed for the core pre-damage phase (EOP) and for the stage involving fuel degradation in the reactor core (SAMG). Thanks to the symptomatic approach to solving emergency conditions, their applicability is not limited by the external conditions. Procedures for the assessment of damage after a seismic event have been executed.

The EOP and SAMG include no mobile or non-technological means and supplies. Possible use of supporting and alternative technical means could be solved operatively by the OER. Documentation for dealing with emergency situations from the EH and TSC is based on the assumption that the ECC or TSC have access to data. However, documentation is not available for the case of the ECC and TSC being activated in different locations.

The personnel in the NPP are sufficiently qualified and trained to use the EOP and SAMG, as well as to assess the damage to the systems after a seismic event. The personnel are also trained to carry out instructions or provide a power supply from on-site or off-site sources in case of an SBO.

No deficiencies were identified in terms of an insufficient number of persons to mitigate the above consequences of beyond design basis events in the IOER or SOER.

In case of the large-scale destruction of the infrastructure and long-term inaccessibility of the location (destroyed buildings, damaged communications etc.) the following problems may be encountered:

- The new personnel might not be able to reach the site in the initial days. In this case, the required activities would have to be carried out by the personnel present on-site at the time of the event. Switching would be coordinated in cooperation with the state administration (IRS, army, etc.).
- Very likely, it would be impossible to use emergency response shelters, emergency headquarters or the technical support centre, which are located under seismically fragile buildings and are not protected against floods. The inaccessibility of the technical support centre could be partially solved by using CR or ECR. The possible inaccessibility of the shelters would be solved operatively by evacuating non-essential personnel from the location.

Unavailability of the power supply

The limited capacity of the accumulator batteries of the Category I SPSS could complicate certain actions and disable certain measuring systems. This can be delayed by a controlled disconnection of non-essential appliances. The scope and sequence of systems and components to be switched off in order to relieve the batteries depends on their importance for the emergency event and strategy used. The goal of reducing the load on the batteries is to extend the operability of the I&C and power supply for the systems necessary for essential safety activities (start of the DG and restoration of the power supply, isolation of routes for transfer of coolant from the I.C, regulation of pressure in the SG and in the I.C, isolation of the containment, etc.).

Until the batteries are completely discharged, the emergency lightening would function. Loss of lighting could complicate the situation for the staff and, therefore, also extend the time necessary to carry out tasks. If the power supply in both units is lost, the shift personnel could be overloaded by activities related to restoring the power supply.

Potential failure of instrumentation

All required information about the state of the components and values of the parameters essential for coping with severe accidents are available in the PAMS and they are either processed directly in the PAMS or sent to other I&C of safety systems.

All systems providing information about safety relevant parameters into the PAMS, as well as the PAMS itself, are qualified for emergency conditions and post-accident conditions and they are supplied from secure power supply systems. Although these systems are qualified for design basis emergency and post-accident conditions, they were designed with a certain consideration of the parameters necessary to cope with severe accidents. For example, the temperature measurement system at the outlet from the reactor core measures temperatures up to 1,300 °C. In addition, the range for hydrogen concentration in the containment is up to 10%.

Only a limited set of parameters is used in diagnosing emergency situations and for verification of the implementation of selected strategies. These parameters are verified using measured values of certain variables from standard instrumentation. For each parameter several variables are specified, which allow indirect verification the parameter (value, trend).

Direct measurement is always used in combination with one or several measurements of alternative values, which can be used for deriving the value or trend of the parameter of interest. In some cases (during severe accidents), the value or trend of the parameter in question cannot be measured directly, either because it is not available or because measurement of the parameter does not exist. Such parameters are quantified using calculations (simple graphs of dependency between parameters). These calculations can be fed as input either the directly measured values, or pre-defined values.

Sensors and converters located inside the containment will, in accordance with the project, be capable of long-term operation under a temperature of 150 °C and overpressure of 0.4 MPa. Subsequent systems processing signals located in the I&C boxes will, in accordance with the design, be capable of operation until the temperature in the rooms reaches 40 °C.

Potential effects from the other neighboring installations on the site, including considerations of restricted availability of trained staff to deal with multi-unit, extended accidents

As near as 900 m from the units of the Temelín NPP, there are three branches of a major gas pipeline. The simultaneous full-diameter rupture of all three gas pipelines with a subsequent discharge of gas would not have a negative impact on the safety systems of the Temelín NPP.

To control possible leaks of natural gas from the pipeline, sensors detecting increased concentrations of natural gas are installed in selected locations of ventilation openings. In case the natural gas were to spread underground (or snow or ice cover) in frozen soil, the increased concentration of natural gas would be detected by measuring probes located in measuring locations at the edges of the power plant towards the pipeline. These sensors feed data to the signaling and related activities if natural gas is detected where defined. An appropriate response for this case has been included in the emergency response system.

Other risks from internal sources have also been evaluated; the warehouse of chemicals, technical gases and diesel fuel management. These were identified as negligible.

III.6.1.4 Conclusion on the adequacy of organizational issues for accident management

Implemented in the Temelín NPP is a system capable of coping with accidents up to the 4th level of defence-in-depth and an emergency response system for the 5th level of defence-in-depth. This functional and interconnected system for coping with emergencies and for emergency response is secured in the Temelín NPP by a sturdy set of personnel, administrative and technical measures. In the personnel area it is: The existence of emergency response organization and functions (positions) within this organization, which carry out specific activities. In the administrative area it is: Processing and implementing the necessary procedures, manuals and instructions using the capacities of technical support centres. And in the technical area it is: Ensuring functionality in the scope necessary for the implementation of strategies. In the first (preventive) phase of an event, actions are taken by the current shift staff. In case the event grows beyond the capacity of this staff, the second phase (mitigating consequences) starts and emergency response organization is activated. In this case, the responsibility for coordinating activities is assumed by emergency headquarters assisted by the technical support centre.

In case of an extraordinary event, all essential activities would be controlled and carried out from places also protected against radioactivity in the air. The TSC and EH, which coordinate strategies according to the SAMG, are located in the ECC – a secure workplace capable of sustaining habitability for at least 72 hours without external support. Remote tasks necessary for the implementation of strategies would be carried out by shift staff from the CR or ECR. Local tasks and possible repairs of systems in clean parts of the reactor chamber, machine room or external objects would be carried out by response groups stationed in the operating support centre.

The concept of coping with technological accidents in the Temelín NPP is based on a symptomatic approach. Solutions to technological accidents implement strategies from the EOP, whose main priority is to restore heat transfer from the reactor core and thus prevent damage to the 1st barrier preventing fission products from escaping (covering of fuel). To mitigate the consequences of severe accidents, strategies from the SAMG are implemented, whose main objective is to prevent damage to the 3rd barrier against radioactive leak (containment), which at that moment is the last undamaged barrier. The EOP and SAMG are updated on a regular basis to include findings from the simulator and from emergency exercises, but also to include external findings. Currently, SAMG are being prepared for shutdown states. These guidelines are designed for situations in which a beyond design basis emergency situation, as a result of fuel degradation in the reactor core, develops into a severe accident with a shut down unit or for beyond design basis emergency situations caused by degraded fuel in the SFSP.

An emergency response system was implemented for situations when the safety of a unit or the entire site is in threatened and for situations that cannot be controlled using just the shift staff. After announcing an emergency event of any degree (Alert, Site Emergency, General Emergency), the emergency response organization is activated. The emergency response organization has an internal component (IOER) consisting of shift staff, and a standby component (SOER), consisting of specialists from the Temelín NPP who are on standby.

The selection of shift employees and employees for the SOER depends on the qualification requirements but other criteria are considered as well, including their level of knowledge and expertise. The readiness of the shift and technical staff to cope with technological accidents is verified on a regular basis in large-scale simulations, which include staff from the TSC, as well as emergency exercises.

The organizational method of coping with extraordinary events (include severe accidents) is specified in the internal emergency plan approved by SÚJB.

In case of emergency conditions (design basis and beyond design basis events without fuel degradation), all currently available means will be used according to their design to meet the requirements of the EOP. The SAMG assume fulfilling the required tasks using all available systems and facilities, i.e. including non-design means.

Available on the site of the Temelín NPP is a fire rescue unit with appropriate fire equipment, trained to cope with fire in any part of the location. The pumping equipment of the LFRU belongs to the main mobile non-technological means that can be used for transporting and pumping media.

The program for coping with accidents within the Temelín NPP is also supported analytically. Analytical support is based on a probabilistic – deterministic approach; the most likely accident scenarios are selected that lead to severe accidents and these are then deterministically analyzed using integral calculation codes. The outcome of analytical support is a set of findings, understanding of events related to severe accidents and their timing, identification of weaknesses of the project, specification of activities that would mitigate the consequences of severe accidents, validation of response activities in case of heavy accident, and identification of the source member for the assessment of the possible radiological impact. A simulator is also available, which displays events occurring in specific severe accident scenarios.

With respect to these facts, the ability to cope with accidents and emergency readiness are found to be comprehensive and sufficient, without major deficiencies.

III.6.1.5 Measures that can be foreseen to enhance accident management capabilities

Emergency measures that secure the 4th level of defence-in-depth and the system of emergency readiness that secures the 5th level of defence-in-depth are designed for the simultaneous failure of all project measures for the 1st, 2nd and 3rd levels of defence-in-depth.

Although there are several diverse systems for implementing each of the strategies designed to cope with accidents, opportunities for further increase of safety were identified in the area of coping with severe accidents. The availability of the proposed additional technical means and implementation of instructions for their use for fulfilling safety functions (in case all project systems are lost) would increase the ability of the Temelín NPP to cope with beyond design basis scenarios before they develop into severe accidents.

- In the area of technical readiness, this includes the adequacy of alternative technical means for the fulfillment of safety functions in case all design SSC are lost. However, previous findings have already led to changes of the project (currently in preparation), which will increase the resilience of the units in case of severe accidents. This is related to the fact that the current project cannot completely exclude the possibility that the integrity of the containment is disrupted by hydrogen produced during a severe accident. Also, there are only limited possibilities of ensuring long-term integrity of the containment after the bottom of the RPV (reactor pressure vessel) is damaged.
- In the area of administrative control, this is, in particular: A manual for coping with severe accidents in case of a shutdown, which have not been finished for the Temelín NPP. The EOP, SEOP and SAMG are maintained (updated) regularly as a part of the “Maintenance Program”.
- In the personnel area, problems may occur with accessibility of the location, which implies potential problems with the usability of the ECC and thus also with the control of activities, decisions about very risky variants when facing accidents and with the communication and warning of the staff.

In order to increase the efficiency of the system for dealing with accidents, the following measures will be executed:

- Organizational aspect for the most efficient use of existing capacities or identification of additional capacities – crisis plans for coping with the foreseeable states of the Temelín NPP (site-wide effects, loss of emergency response control centres, loss of communication and warning systems, decisions about risky variants of solutions, switching of staff, extreme natural conditions, etc.).
- Finishing some technological instructions/procedures/manuals for coping with selected states and severe accidents within the Temelín NPP (SAMG for shutdown states, SAMG for damaged fuel in the SFSP, EDMG, etc.) with the aim of ensuring cooling and heat transfer from the reactor core and the SFSP and thus preventing radioactive leaks.
- Additional technical measures to secure non-technological support functions (access to buildings, accessibility of fire equipment, safety of emergency control centre and shelters, etc.).
- Alternative means for long-term functioning of the communication channels between all components of the emergency response system (internal and external).

The following table contains opportunities to improve the defence-in-depth in case of events that could lead to a loss of safety functions. The table contains areas that require additional analysis, which were not available at the time this assessment was executed.

Some measures (labeled in the note as “finding of the PSR”) would be implemented even without this targeted evaluation, whose conclusions have confirmed the effectivity and correctness of previous decisions regarding the implementation of measures leading to increased robustness of the original design.

Tab. 36: Opportunities to improve the defence-in-depth in case of events that could lead to a loss of safety functions

Improvement opportunity	**Corrective measure**	**Time period (Short-term I / Medium-term II)**	**Note**
Technical means	Alternative supply of water into the containment reservoir	II	Finding of the PSR
Technical means	System for the liquidationl of hydrogen in the containment in case of a severe accident	II	Finding of the PSR ZKZ B462
Analyses	Localization of melt outside the RPV	II	Finding of the PSR Will be implemented in cooperation with other operators of VVER
Technical means	Verification of the system functions in beyond design basis operating states	II	
Analyses	Analysis of the radiation situation in the CR/ ECR in case of a severe accident		
Rules	Execute “SAMG shutdown” (fuel degradation with open reactor/in SFSP)	II	• Finding of the PSR • Will be started in 2012
Rules	EDMG manuals for the use of alternative means	II	
Staff	Appointing qualified and well-trained staff to the ERO	I	
Emergency response	Ability of the ERO to function outside the ECC	I	
Emergency response	Prepare agreements with external organisations (IRS, army) close to the NPP. Organizational measures.	II	

III.6.2 Accident management measures in place at the various stages of a scenario of loss of the core cooling function

III.6.2.1 Before fuel degradation in the reactor pressure vessel (including last resorts to prevent fuel damage)

Measures for coping with accidents in case of a loss of cooling (before the fuel is damaged) are described in the EOP and there is also a sufficient number of technical means for implementing these measures.

Failure to cool the reactor core is identified by a high temperature at the outlet from the reactor core, with a limiting value of 370 °C. This is the critical temperature for water, with the steam coming from the core zone being near the state of overheated steam. This is a clear symptom indicating that the reactor core was uncovered. If cooling in the reactor core is not restored, the temperature at the output from the active zone will exceed 650 °C. An analysis of the conditions connected with insufficient cooling of the core demonstrated that at the time this temperature is reached, at least three-quarters of the reactor core are uncovered. After the reactor core is uncovered, the heat is transferred by the remaining coolant, but mostly by the steam, which gradually heats up as it flows around the fuel rods. The uncovering of three-quarters of the core will lead to a decrease in the production of steam in the core, which will in turn lead to a decrease in the amount of coolant returning to the core after condensation in the SG. Consequently, the rate of the temperature increase grows. Further increase in the temperature of the overheated steam in the core would create conditions for intense development of steam-zirconium reaction, which represents another source of heat. In this situation, the first barrier against the release of fission products would be in danger.

The strategies for dealing with accidents involving loss of cooling in the core are described in the EOP (phase before the fuel in the core is seriously damaged) and in the SAMG (phase after the fuel in the core has been seriously damaged). A thorough symptomatic approach of the SAMG is expedient for achieving the primary objective – protecting the containment from damage.

Cooling of the core in the phase before the fuel is seriously damaged can be restored by carrying out the activities described in the EOP. The following strategies are defined for restoring cooling in the reactor core:

- Restoring high-pressure makeup for the I.C (high-pressure emergency supply, emergency boroning, normal makeup) to restore the core cooling function.
- Depressurizing of the SG – increasing heat transfer from the I.C to the SG; this leads to a condensation of steam on the primary side of the tubes in the SG. Because the rate of condensation exceeds the rate of steam generation, the pressure in the I.C will start to decrease. This depressurizing of the I.C can be carried out by emptying the hydro accumulator into the I.C.
- Depressurizing of the I.C – reducing pressure in the I.C can ensure a supply of coolant into the I.C from low-pressure sources. Although this method is efficient, the disadvantage is a further decrease in the already low amount of coolant in the I.C if the purpose of depressurizing (to activate low-pressure systems) is not achieved. In any case, depressurizing allows bringing cold coolant from the hydro accumulator.
- Using the residual amount of coolant in the I.C – equalizing levels between the core and the lower mixing chamber by connecting the upper parts of the I.C and cold branches of circulation loops, attempting to activate the main circulation pump (MCP) for possible cooling of the core by bringing water from the hydro valve in the part of the cold branch near the suction opening of the MCP. Activation of the MCP will allow the overheated steam to circulate.

III.6.2.2 After the occurrence of fuel damage in the reactor pressure vessel

Measures for coping with accidents after severe damage to the fuel are described in the SAMG strategies, which use all available means of I.C makeup for restoring the cooling function in the core. Each individual system for I.C makeup is able to supply a sufficient amount of cooling for the transfer of residual heat from the damaged fuel, although flooding of the RPV does not guarantee that the core will be cooled – the core could be in a state when cooling is no longer possible, due to melting.

All strategies are based on the principle of cooling the damaged fuel from the inside of the RPV, i.e. by the addition of water into the I.C. With respect to the thermal output of the reactor and due to the design of the concrete reactor shaft, the RPV in the VVER 1000 units with V320 reactors cannot be cooled from the outside.

As long as the core is intact and flooded with water, the cooling is sufficient to prevent damage to the core. If the reactor core does not contain water, residual heat is absorbed by the materials in the reactor core. If water makeup into the I.C has not been initiated, the reactor core continues to heat up.

Cooling of the core in the phase after the fuel has been severely damaged can be restored by the methods described in SAMG. The following strategies were defined for restoring cooling in the reactor core.

- Supplying water into the hot, dry reactor core, which always has a positive impact on the development of an accident. The method for restoring water supply for the I.C should be optimal in the sense of minimizing the subsequent release of fission products into the atmosphere. If the amount of water supplied is sufficient for removing heat faster than it is being generated, it is also possible to restore the cooling system in the core of the reactor.
- Another method that can be used after the fuel has been severely damaged is depressurizing of the I.C. The goal of depressurizing the I.C is to reduce the pressure in the I.C below a value that cannot lead to direct heating of the containment, because melt will not be expelled from the reactor under high pressure. There are several methods for depressurizing the I.C (system for emergency bleeding of the I.C, release valve of the pressurirser normal injection in the pressuriser, depressurizing of the SG etc.).

III.6.2.3 After a failure of the reactor pressure vessel

After the RPV fails, the debris of the reactor core will end up in the reactor's concrete shaft or in other parts of the containment. If the containment contains no water, the debris of the reactor core will start interacting with the concrete bottom of the containment, a process referred to as Molten Core Concrete Interaction (MCCI). Hydrogen and other incondensable gases will be created during this process. Measures for coping with accidents after the fuel has been severely damaged and melt has fallen to the bottom of the containment are described in the SAMG strategies, which use all available makeup means to cool the melt.

All strategies for cooling the melt that has fallen to the bottom of the containment are based on the principle of pouring water on the melt from above. Flooding the debris of the reactor core outside the RPV will transfer heat from this debris and thus reduce the speed of interaction with the concrete.

One of the outputs from the analyses of sequences leading to severe accidents, selected on the basis of the PSA2 outcomes, is also the time until the integrity of the RPV is disrupted by interaction with the melt from the reactor core. This time period could be about 4.5 hours in the worst-case scenario, assuming that all methods of supplying coolant into the RPV have failed. At this moment, the in-vessel phase of a severe accident changes into the ex-vessel phase, with all related events within the containment (interaction of the melt with the concrete producing hydrogen, direct heating of the containment, etc.).

Damage to the RPV after severe damage to the fuel, which marks the end of the in-vessel phase of a severe accident, is a “cliff-edge” event in terms of negative processes in the containment.
As a preventive measure in case of a severe accident, water is supplied into the containment. The corresponding strategy in the SAMG provides instructions for flooding the containment with water up to the maximum possible level, which serves two purposes: It protects the concrete at the bottom of the containment in case debris of the reactor core releases from the RPV into the containment, and it effectively washes away fission products escaping from the melt.
If the mix of the reactor core debris and concrete is flooded with water, the transfer of heat from the top surface of this mixture will be much more efficient. Water will also leak into cracks between the fragments on the surface of the crust, which further improves the transfer of heat from the pool containing melt of the reactor core.
Supplying water into the containment during a severe accident brings several expected benefits:

- The water at the bottom of the containment will transfer heat away from the reactor core debris and thus limit the interaction between the melt and concrete.
- Water from the sump in the containment can be used to operate the pumps of the emergency makeup and the containment’s spray system.
- Fission products escaping from the reactor core debris at the bottom of the containment will be washed away by the water.

During a severe accident standard methods of makeup can also be used, such as makeup from the supply water reservoirs for refueling or from reservoirs with impure condensate. Furthermore, alternative methods of supplying water can be used, e.g. using stable fire pumps or overfilling the condenser tank.

III.6.3 Maintaining the containment integrity after the occurrence of significant fuel damage (up to core meltdown) in the reactor core

III.6.3.1 Elimination of fuel damage/meltdown in high pressure

Design provisions

The main design tools for depressurizing the I.C: are the system of emergency bleeding of the I.C, the relieve valve of pressuriser), the normal injection system of the pressuriser, and depressurizing of the SG. However, there are also limited possibilities for using these even in beyond design basis accidents.

Depressurizing the I.C can be used as a preventive measure to prevent damage to the fuel in case of high pressure in reaction to insufficient cooling of the I.C in the EOP. If all these methods of reducing pressure in the I.C fail, the SAMG contains instructions for assessing the availability of systems necessary for depressurizing the I.C and the negative consequences of this depressurizing. The goal of depressurizing the I.C is the reduction of pressure in the I.C below a value that has been demonstrated as limiting, i.e. below this temperature the containment cannot be heated directly, because melt is not released from the reactor under high pressure.

Operational provisions

The reduction of pressure in the I.C is one of the highest priorities when coping with severe accidents. There are several methods of depressurizing the I.C:

- Using the system for emergency bleeding of the I.C.
- Pressuriser relieve valve.

- Normal injection into the pressuriser.
- Depressurizing the SG.

The reduction of pressure in the I.C during a severe accident has several positive consequences:

- A lower probability that melt would release under high pressure (High Pressure Melt Ejection)
- A lower probability of creep to the pipes in the SG and primary SG collectors (lower probability of leaks outside the containment)
- Possibility of makeup the I.C from more sources (especially low-pressure).

III.6.3.2 Management of hydrogen risks inside the containment

Design provisions, including consideration of adequacy in view of the hydrogen production rate and amount

The integrity of the containment is threatened most by two types of hydrogen burning – quick deflagration and the change from quick deflagration to detonation. To assess the hydrogen risk the time trends of the spread and distribution of hydrogen, produced in case of severe accidents all around the containment, were analyzed.

The containments of units in the Temelín NPP are equipped with a post-accident hydrogen liquidation system, designed for design basis accidents. This system contains passive autocatalytic recombiners and it is able to dispose of hydrogen released during accidents for a long period of time, thus keeping the concentration of hydrogen low enough to prevent its combustion – but only during design basis accidents. The existing hydrogen management system might not be sufficient for severe accidents. However, a project is currently being prepared that involves the installation of a hydrogen management system to liquidate hydrogen produced during severe accidents.

A potential method for reducing the amount of hydrogen in the containment is venting of the containment (filtered or unfiltered), which is only possible by employing systems that were not designed for this purpose. This possibility has not been analyzed yet.

Operational provisions

The existing measures for coping with accidents threatening the integrity of the containment by hydrogen are described in the SAMG strategies, which employ all available means to prevent dangerous forms of hydrogen burning. Apart from the strategy of intentional burning of hydrogen, which has only limited usability and is based on the random ignition of hydrogen using one of the electrical appliances in the containment, all other strategies for preventing dangerous forms of hydrogen burning rely on creating conditions that suppress the combustibility of hydrogen. These strategies, therefore, do not lead to the actual liquidation of hydrogen – only to restraining the conditions for it to burn. Using all existing options, therefore, does not eliminate the possible hydrogen threat to the containment completely; however, it provides more time for the recombiners of the current system.

The last resort preventing damage to the integrity of the containment described in the strategies implemented in the SAMG is controlled venting of the containment using systems that were not, by design, intended for venting. The SAMG strategies also deal with the possible negative consequences of a hydrogen leak outside the containment.

III.6.3.3 Prevention of overpressure of the containment

Design provisions, including means to restrict radioactive releases if the prevention of overpressure requires steam/gas relief from the containment

The design function of the containment is to limit the potential radiation consequences of an accident in the reactor. The structure of the containment was designed with this purpose in mind; to prevent leaks outside the containment to very low levels even with high internal overpressure in the containment. Because the containment contains the entire pressure boundary of the I.C, the containment is the last barrier preventing the release of radionuclides that could be released from fuel or coolant of the I.C in case of an accident.

The integrity of the containments within the Temelín NPP is ensured by the following systems:

- Containment isolation system – separating valves automatically closed when the pressure in the containment increases. Operability depends on power supply.
- System for pressure reduction in the containment – spray pumps and supply tanks with chemical reagents capturing post-accident iodine. The operability depends on the power supply.
- Post-accident hydrogen management system – passive auto-catalytic recombiners, designed for design basis accidents – does not require a power supply.

The systems for pressure reduction include 3 divisions of spray systems, each of which is capable of reducing the pressure in the containment by condensing steam generated from a rupture in the steam piping up to the scope of the primary circulation piping (3 x 100%). The design load on the containment in case of a large-scale LOCA is less than 150^{o}C and 0.49 MPa.

Overpressuring of the containment is one of the leading causes of damage to the integrity of the containment that could potentially lead to the release of fission productions into the atmosphere. Ovepressure in the containment during a severe accident could also be a consequence of dynamic processes (burning of hydrogen) or a long-term accumulation of steam or incondensable gases within the containment. Dynamic events could lead to pressure peaks, which might not be mitigated by the normal transfer of heat from the containment (i.e. the energy increase in the containment is greater than the transfer capacity of spray systems).

Analyses have been carried out aimed at determining the limiting pressure increase in the containment. The conclusion of these analyses is that, until the RPV melts through and the melt leaks to the bottom of the concrete shaft of the reactor, the pressure in the containment cannot reach levels sufficient to pose a serious threat to its integrity. Only after the melt starts interacting with the concrete in the ex-vessel phase, could the pressure increase above the levels threatening its integrity.

Strength calculations for the containment have shown that after the design pressure in the containment is exceeded, the structure of the containment behaves linearly. Only after cracks appear in the concrete of the inside wall does the steel lining become gradually plastics-like and eventually the containment loses hermeticity.

Reaching a pressure value sufficient for damaging the integrity of the containment (about 1.6 times the design basis pressure) is a “cliff-edge” event in terms of the threat it poses to the integrity of the containment as a result of overpressure.

Operational and organizational provisions

The EOP assumes that the spray system of the containment will be used in such a way that the pressure inside the containment remains within the limits of the design basis. The containment spray system is capable of ensuring, at least, transfer of residual heat in case of an accident. Therefore, the spray system should, in the long run, maintain pressure inside the containment at a level that corresponds to the pressure in the surrounding atmosphere, provided that there has been no significant release of incondensable gases from the interaction of the melt with the concrete.
In the phase before the fuel in the reactor core is significantly damaged, as well as after it has been significantly damaged, the proposed strategies are designed to prevent a bypass of the containment and to minimize the release of radioactive substances via a damaged heat-exchanging surface in the SG or via the ECCS (safety injection system) to the ESW.
Measures designed for coping with accidents threatening the integrity of the containment by high pressure are described in the SAMG strategies, which employ all available means to reduce pressure in the containment. The corresponding SAMG strategies provide instructions for preventive measures leading to reduced pressure in the containment in case its integrity is threatened by overpressure:

- Spray systems in the containment (standard spray system in the containment or fire water)
- Injecting containment using fire pumps is an alternative method (assuming there is a power supply). The locations of the fire pumps and their tanks ensure sufficient diversification when compared with containment spray systems.
- Ventilation and venting units in the containment (with coolers)
- Various (non-design) routes for filtered or unfiltered venting.

Venting the containment employs systems that were not intended for venting by design. It is one of the possible methods of mitigating a serious threat to the containment by high pressure. Venting the containment reduces pressure in the containment, but leads to the release of fission products into the atmosphere and therefore, it is considered to be the last-resort option to prevent containment failure.

III.6.3.4 Prevention of re-criticality

Design provisions

In case water with a low content of H_3BO_3 is being supplied to the I.C and thus the concentration of H_3BO_3 in the I.C decreases, it is possible that repeated criticality is reached, and the output of the reactor is increased as a consequence of increased moderation of neutrons (provided that the original geometry of the reactor core is preserved). If the original geometry of the reactor core has been lost, and therefore the ability of neutron moderation is reduced, the system cannot reach a critical state. The possible return of the reactor to operation at power does not pose an immediate threat, because the output is limited by bubbles being created in the reactor core.
After the fuel has been severely damaged and the geometry of controlling elements in the reactor core has been lost, a cavity could be formed with no absorbers. Non-existence of a moderator (water in these conditions always turns into steam) will lead to a loss of moderation capacity. With respect to the damaged geometry of the reactor core, a critical state cannot be reached in a large volume.

Operational provisions

Measures for preventing the decrease of boron concentration have a higher priority for activities according to the EOP in the preventive phase before the fuel is damaged while the reactor core still has its original geometry, which allows the moderation of neutrons and a critical state. The SAMG contain instructions for measures to be taken in the phase after fuel has been seriously damaged and the original geometry of the core has been changed, if the rate of output increase grows. However, because the geometry of the reactor core is changed, it is impossible for a critical state to be reached in the large volume of the core.

III.6.3.5 Prevention of basement melt-through

Potential design arrangements for retention of the corium in the pressure vessel

With respect to the thermal output of the reactor and the design of the VVER 1000 reactor concrete shaft units with V320 reactors, the RPV cannot be cooled from the outside.

Potential arrangements to cool the corium inside the containment after a reactor pressure vessel rupture

After the RPV fails, debris from the reactor core could be moved to the concrete shaft of the reactor or other parts of the containment.
Measures for coping with accidents when the integrity of the containment is threatened by an interaction of the melt with the concrete are described in the SAMG strategies. As a preventive measure, water is supplied to the containment in case of a severe accident. The corresponding SAMG strategy provides instructions for flooding the containment with water up to the maximum measurable level, which not only provides protection to the bottom of the containment in case debris of the core from the RPV reaches the containment, but also washes away fission products escaping from the melt.
Analyses that have been carried out have shown that cooling the pool with melted material in the shaft using water can reduce the speed of decomposition of the concrete and thus delay failure of the containment until a late phase of an accident.
Flooding debris of the core outside the RPV using water would ensure the transfer of heat from the debris and thus reduce the speed of degradation of the concrete. During a severe accident, the following strategies are employed to supply water into the containment:

- Supply of water from the supply tanks for refueling.
- Supply of water from the tanks containing impure condensate.
- Supply using fire pumps.
- Overfilling of the condensation tank.
- If the RPV is damaged, water added to the I.C will also flow into the concrete shaft of the reactor through the RPV opening, which will help cool the debris from the core in the reactor shaft.

Mitigating the degradation of the containment’s concrete bottom will delay or completely stop potentially massive release of radioactive substances into the environment after the bottom of the containment melts through. These measures also prevent a large-scale steam explosion and by reducing the thickness of the melt, also increase the probability that it will be cooled and the degradation of concrete by melted material would be stopped completely.

To increase the sturdiness of the containment bottom in the concrete shaft of the reactor against interaction with the melt after the RPV has failed, a modification was implemented that involves clogging of the ex-core channels of neutron flow measurement, which leads across the bottom of the containment. The channels have been filled with removable steel cases filled with heat-resistant material. This solution ensures high resilience against melt with no impact on the instrumentation of neutron flow measurement.

Cliff edge effects related to the time delay between reactor shutdown and core meltdown

As follows from analyses of SBO scenarios that lead to a loss of heat transfer from the I.C by the SG, without carrying out alternative activities as described in the EOP, there is a very short time span during which heat transfer from the I.C must be restored. In the worst-case scenario the temperature at the outlet from the reactor core, 650 °C, could be reached in about 2.5 or 3.5 hours after the SBO event. The temperature at the outlet from the reactor core increasing past the 650 °C limit is a “cliff-edge” event in terms of severe damage to the fuel in the reactor zone.

In case of a long-term loss of cooling for the reactor core, the integrity of the RPV could be damaged by melt from the core. The time for this worst-case scenario to materialize, assuming that all methods of supplying coolant into the RPV tank have failed, is about 4.5 hours. This moment marks the end of the in-vessel phase of a severe accident and the start of the ex-vessel phase.

After melt has been moved from the RPV to bottom of the containment, it starts interacting with the concrete. One result of this interaction is the disintegration of concrete, combined with the intense production of hydrogen. This would weaken the bottom of the containment and once the bottom is weak enough so that the weight of the melt breaks through it, the melt releases into the bottom, non-hermetic part of the reactor building. In the worst-case scenario, assuming all methods of cooling have failed, the melt could penetrate into the bottom (non-hermetic) part of the reactor building about 24 hours after an accident.

III.6.3.6 Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

Design provisions

The integrity of containments in the Temelín NPP is, by design, ensured by the following systems:

- Containment isolation system – separating valves automatically close when the pressure in the containment increases. The operability depends on the power supply.
- System for pressure reduction in the containment – spray pumps and supply tanks with chemical reagents capturing post-accident iodine. The operability depends on the power supply.
- Post-accident hydrogen management system – passive auto-catalytic recombiners, designed for design basis accidents – does not require a power supply.

Operational provisions

Most of the strategies described in the SAMG (supplying water into the containment, transfer of heat, maintaining pressure in the containment) require a power supply for successful implementation.

III.6.3.7 Measuring and control instrumentation needed for protecting containment integrity

The basic measurement for the strategies dealing with threats to the integrity of the containment due to high pressure is the measurement of pressure in the containment. Redundant pressure measurements in the containment are carried out in the PRPS and communicated via the PAMS. The PRPS and PAMS systems are qualified for emergency conditions and post-accident conditions and they are supplied from the accumulator batteries of the secure safety systems. Although these systems are qualified for a design basis emergency and post-accident conditions, they were also designed considering to a certain extent their performance in severe accidents (range of pressure measurement in the containment up to 1.6 MPa). To verify the success of a strategy to mitigate high pressure as a serious threat, the SAMG identifies parameters whose values are to be communicated via the PAMS.

The containments of the Temelín NPP units contain a system of post-accident measurement of hydrogen concentrations (PACHMS – Post-Accident Containment Hydrogen Monitoring System). This measurement is a part of the PAMS and its operability depends on the operability of the PAMS. In total, 16 sensors are located in the containment, covering all rooms in which hydrogen could accumulate. The PACHMS system measures the concentration of hydrogen within the range of 0 ÷ 10 %.

No active functions depend on the post-accident containment hydrogen monitoring system, and the system for liquidating post-accident hydrogen (in case of design basis accidents) also burns hydrogen independently of this measurement. The outputs from the post-accident hydrogen monitoring system are used for coping with the accident (diagnostics of the emergency, selection of a strategy for controlling the combustibility of hydrogen, assessment of the negative consequences of burning hydrogen etc.).

The PACHMS, as well as PAMS, are qualified for emergency and post-accident conditions and they are supplied from the accumulator batteries of the safety supply systems. Although these systems are qualified for a design basis emergency and post-accident conditions, their design includes the considerations of requirements during severe accidents.

Although the post-accident measurement of hydrogen concentration in the containment provides an important input for coping with emergencies, the amount of hydrogen produced can be estimated using a tool from the SAMG based on the development of an emergency situation even without knowing the measured values.

Temperatures at the outlet from the core are used to evaluate loss of cooling in the core. The temperature at the outlet from the core is measured using thermocouples with a range up to 1,300 °C. Measuring is a part of the PAMS, and it is qualified for emergency and post-accident conditions. Temperature measurement at the outlet from the core is sufficiently redundant; 95 thermocouples are located at the outlet from the core and an additional 3 thermocouples are located under the lid of the reactor. The temperature of the cold ends of the thermocouples is measured using redundant resistance thermometers.

There is no direct measurement for the assessment of damage to the RPV and for the amount of melt that has leaked into the containment. In this phase of an accident, the events can be monitored only indirectly, based on the values of the parameters of the environment inside the containment. The success of the strategy of flooding the containment can be verified by measuring the level in the containment – either by measuring the level in an impure part of the containment reservoir, or by measuring the level in the containment. To verify the success of a strategy for flooding the containment, the SAMG identifies parameters to be communicated by the PAMS.

There is no direct measurement for the quantification of the interaction between the melt and the concrete at the bottom of the containment. In this phase of an accident, the events can be monitored only indirectly, based on the values of parameters of the environment inside the containment. The values of these parameters are available in the PAMS. This includes: Measuring pressure, temperature, level and hydrogen concentration in the containment and measuring pressure in the I.C. Using the values from these measurements and instructions from the SAMG, it is possible to determine indirectly whether interaction between the melt and concrete is possible or unlikely under given conditions.

The basic measurement in case of a large-scale release of fission products is the dosage output and activity. In order to measure the dosage output and activity, it is possible to use dosage output measurements from inside and outside the containment, dosage outputs and activity measurements in the chimneys, and measurements from teledosimetric system fixed on the fence around the power plant. The ranges of all these measurements are sufficient for operating, emergency and post-accident conditions.

Radiation measurements with ranges sufficient for emergency and post-accident conditions are carried out in the PAMS, and all radiation measurements are also communicated via this system. Although these systems are qualified for design basis emergency and post-accident conditions, their design included considerations of the requirements during severe accidents (e.g. the measurement range of the dosage output in the containment is up to 10^5 Gy/h).

III.6.3.8 Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

Both units of the Temelín NPP are technologically independent and constructionally separate. Accident-related activities are controlled for both units from the ECC (TSC and EH), and response actions within units are carried out by the staff in the unit. Depending on the current situation on each unit, the capacities can be operatively moved between the units. In case of an accident in one unit, the TSC staff has instructions for making the decision on how the other unit will be run. In case the situation escalates into a severe accident in both units, the same SAMG will be used for both; however, the situation in each unit would be assessed independently and the TSC and EH would coordinate the activities.

III.6.3.9 Conclusion on the adequacy of severe accident management systems for protection of the containment integrity

In order to cope with beyond design basis and severe accidents, all available technical means are used; i.e. including those that are not primarily designed for coping with severe accidents. Use of this means is described in the corresponding strategies in the EOP and SAMG. The strategies are designed to succeed, i.e. one of the secondary objectives of the SAMG is to restore the operability of systems and facilities to the greatest possible extent. Implementation of the strategy in any described manner leads to success. Success in this context means fulfillment of the main objectives of the SAMG, i.e. bringing the unit to a controlled stabilized state and reducing the release of radioactive substances.

There are several diverse systems for the implementation of each SAMG strategy. Analytical validation of the SAMG has proven that some strategies are not fully supported by the corresponding technical means; nevertheless, implementation of the proposed strategies in response to threats caused by severe accidents using existing systems and facilities that are not by design intended for severe accidents, leads to successful coping with severe accidents.

Long-term post-accident activities related to coping with severe accidents focus on the continuation of heat transfer and the elimination of high-energy events (burning or hydrogen explosion, etc.), depending on the situation in the unit. In this case it could be very difficult to define precisely what condition the unit is currently in and thus define possible threats. However, after bringing the unit back to a controlled and stable state, the basic conditions of ending the SAMG are fulfilled. Before ending the SAMG and continuing with long-term post-accident activities, the SAMG specifies the exact methods for the identification of the state of the unit, scope of the damage and long-term risks.

Long-term post-accident activities are shifted from the phase of “searching for suitable measures” to the phase of “ensuring the long-term functionality of found and successfully applied measures”, e.g. ensuring that alternative water supplies do not fail (loss of power supply, running out of water, failure of components). This is, therefore, related to searching for alternatives to already successfully implemented measures, i.e. searching for other measures that would immediately replace the currently implemented measures or that can serve as backup in case the currently implemented measures are lost.

Assessment of the measures protecting the integrity of the containment for certain highly improbable, beyond project basis scenarios has revealed the possibility that radioactive substances will be released into the environment as a consequence of a hydrogen threat to the containment integrity in case of a severe accident and limited means for prevention of the containment integrity damage due to a melted containment bottom.

III.6.3.10 Measures that can be foreseen to enhance capability to maintain containment integrity after the occurrence of severe fuel damage

In spite of very strict measures for coping with accidents designed to prevent a loss of containment integrity that can be found in the SAMG, certain opportunities have been found for increasing the ability to maintain integrity of the containment after severe fuel damage. These opportunities lie in designing and implementing additional means for ensuring integrity of the containment (i.e. preventing a leak of fission products) in case of a severe accident. These additional means include a hydrogen liquidation system in the containment and measures for the localization of melt at the bottom of the containment.

The opportunities to improve defence-in-depth protection in case of events that could lead to a severe accident can be found in the following table. The table contains also areas that require the execution of additional analyses, which were not available at the time this assessment was elaborated.

Some measures (labeled in the note as “finding of the PSR”) would be implemented even without this targeted evaluation, whose conclusions confirmed the effectivity and correctness of the previously made decisions regarding the implementation of measures leading to increased strength of the original project.

Tab. 37: The opportunities to improve defence-in-depth protection in case of events that could lead to a severe accident (after the occurence of severe fuel damage)

Improvement opportunity	Corrective measure	Time period (Short-term I / Medium-term II)	Note
Technical means	Hydrogen disposal system in the containment for severe accidents	II	Finding of the PSR ZKZ B462
Analysis	Localization of melt outside the RPV	II	Finding of the PSR Will be addressed in coordination with other operators of VVER1000

III.6.4 Accident management measures to restrict radioactive release

III.6.4.1 Radioactive release after loss of containment integrity

Design provisions

The SAMG strategies aim to prevent the large-scale release of fission products. When all strategies designed to prevent a loss of containment integrity fail, it is time to implement emergency response measures.

Large-scale leakage is leakage exceeding 100 micro Gy/h (criterion for announcing a 3rd-degree extraordinary radiation event according to the Internal emergency plant). Leakage of fission products of this magnitude indicates that all barriers have been breached, including the containment (or its bypass).

There could be one or several sources of leakage (containment, SG, non-hermetical rooms, ESW).

In the assessment of the impact of radioactive leaks, all biological effects of ionizing radiation have been considered. The specification of limiting values in terms of biological effects of radioactive leaks is beyond the scope of this assessment.

Operational provisions

The identification of large-scale leakage of fission products requires the consideration of all four sources. The activities carried out to reduce leakage from one source could have an impact on the leakage from another source. The strategy focuses on the identification of the place of the leakage to make sure that activities carried out according to this manual will not worsen the leakage. Preventing the leakage of fission products into the environment and preventing a loss of containment integrity (as the last barrier against the leakage of fission products) are the main goals of the SAMG. However, the SAMG also contains strategies for stopping, or at least reducing, the leakage of fission products after containment integrity has been lost. These strategies employ all available means. If the containment integrity is lost and the large-scale leakage of fission products is ongoing, it is necessary to immediately implement a strategy for their mitigation, because this is the strategy with the highest priority within the SAMG.

Measures designed for coping with accidents after containment integrity has been lost are described in the SAMG strategies. These strategies are implemented by shift employees from the CR or ECR. In case the CR and ECR are uninhabitable and large amounts of radioactive material is escaping from the containment, implementing SAMG strategies could be hindered by lack of control over the components of the safety systems.

The leakage of fission products can be reduced by the containment spray system or cooling venting systems in the containment by washing and reducing pressure. The last-resort method for reducing the pressure in the containment and limiting the amount of escaping fission products is filtered venting. Before using filtered venting, it must be verified that this method will not worsen the situation.

In terms of the protection of staff and the population, the SOER creates a quick mobile monitoring group, which monitors and evaluates radiation in the affected sectors. Preventive measures designed to protect the population are assisted by RTARC.

III.6.4.2 Accident management after uncovering of the top of fuel in the fuel pool

The SFSP is located in containments of the units of the Temelín NPP. If spent fuel is stored in the SFSP, it must also contain a sufficient amount of coolant and heat must be transferred out of the SFSP. When in storage mode, the level in the SFSP must exceed 792 cm. If the level in the SFSP drops below 550 cm, the tops of the fuel rods will be uncovered.

In case standard cooling of the SFSP fails (whether as a consequence of a low level or after heat transfer failure) the water in the SFSP is makeup using a containment injection system with settings for emergency supply of the SFSP. This system supplies coolant into the SFSP, which then drains from the SFSP to the bottom of the containment and subsequently into the containment reservoir, which ensures the backup transfer of heat from the spent fuel in the SFSP via the ECCS exchanger. This cooling circuit is independent of the SFSP cooling system and offers an alternative method of heat transfer from the spent fuel stored in the SFSP. The heat from the SFSP is transferred by vaporization into the containment and vaporization is compensated by a makeup system – the containment spray system. In case the heat cannot be transferred from the spent fuel in the SFSP via the ESW to the ultimate heat sink, this method of long-term cooling is limited to the passive thermal capacity of the containment.

A backup method of transferring heat from the SFSP is, in case normal cooling is lost, the emergency the SFSP cooling employing containment spray system.

Loss of cooling in an SFSP containing spent fuel has been analyzed quantitatively. The results of the calculations are the time reserves before saturation temperature is reached and the tops of the stored fuel assemblies are revealed after the SFSP cooling system is lost.

The results of the calculations of heating trends and time reserve until boiling point depend on many factors, such as the number of fuel assemblies in individual sections of the SFSP (thermal output), the time since the fuel sets were moved from the core, the level in the SFSP at the moment the heat transfer function was lost, the initial temperatures in the SFSP etc. Based on analyses performed, it can be said that, depending on the initial conditions, the trend of temperature increase in the SFSP after cooling has been interrupted is between a few °C/hr to several dozen °C/hr and the time until boiling ranges from several hours to several dozens of hours. With a maximum thermal load in the SFSP, the loss of heat transfer from the SFSP will not lead to damage of the stored fuel assemblies until the late phase of an accident.

The time until the tops of stored fuel sets are revealed is a “cliff-edge” threat to the cooling of the spent fuel stored in the SFSP.

In terms of the time necessary to restore cooling of the spent fuel stored in the SFSP, the situation is better than a case when heat transfer from the core is lost; however, long-term loss of heat transfer function, exceeding several dozens of hours, without makeup water using a backup method, could damage the spent fuel stored in the SFSP.

Hydrogen management

A later phase of cooling function loss in the SFSP involves the vaporization of water, uncovering of the stored fuel and steam-zirconium reaction in the upper part of the fuel rods. After the fuel is uncovered it would, when exposed to air, overheat and oxidize, leading to much more extreme thermal escalation (heat released during oxidation by oxygen is higher than for steam), nitration in the upper part if oxygen is missing and subsequent oxidation after spending unoxidized Zr in the lower part.

These sources of hydrogen are located in the containment. Assuming that the routes leading through the walls of the containment are isolated, it is practically impossible that hydrogen would release from the SFSP outside the containment and, therefore, the hydrogen threat to the key parts of the unit outside the containment can be excluded.

Providing adequate shielding against radiation

In order to shield the radiation from spent fuel sets, the water level cannot drop below 783 cm. The location of the SFSP inside the containment means that even if the water level in the SFSP drops below the level necessary for shielding, the radiation from spent fuel will not reach any persons. Under normal operation of the unit (in Modes 1 through 4) the containment is closed and persons may only enter with a special permit (which includes assessment of radiation).

Because in Modes 5 and 6 the hermetic openings on the containment could be open and persons may be in the containment, carrying out tasks related to shutdown, this situation could lead to a health risk for these persons. Therefore, one of the measures carried out immediately after an emergency has been detected is the evacuation of all persons from the containment and closing it hermetically.

Restricting releases after severe damage to spent fuel in the fuel storage pools

The technical means for mitigating the consequences of damage to the fuel in the SFSP are based on the continuation of water makeup and transfer of heat into the containment. Possible leakage from the SFSP is to be isolated according to the EOP. The SAMG for shutdown modes in case of accidents, which include melting of the fuel in the SFSP, have not been executed yet.

No analyses have been carried out for situations involving damage to the spent fuel stored in the SFSP. Because there is an alternative way of makeup the SFSP using the containment spray system, long-term loss of heat transfer from the SFSP (without simultaneous loss of heat transfer from the core) is not considered.

With respect to the fact that the SFSP is located inside the containment, if both heat transfer functions (from the SFSP and from the core) are lost simultaneously, the limitations following from the loss of heat transfer from the core are “cliff-edge” conditions, because in terms of the time necessary for restoring cooling of the spent fuel stored in the SFSP, the situation is better than in a case when the heat transfer function from the core is lost.

Instrumentation needed to monitor the spent fuel state and to manage the accident

The key parameter for assessing the loss of heat transfer from spent fuel stored in the SFSP is the water level in the SFSP. As long as the fuel is covered by water (even boiling), residual heat is being transferred from the fuel. Once water boils away from the SFSP and the tops of the fuel rods are uncovered, the fuel systems will start overheating. The level in the SFSP and several other parameters, such as the condition of the ESW system and flow rate from the ESW system into the exchanger for SFSP cooling, are communicated via the PAMS. Also, the temperature is measured in the SFSP.

Availability and habitability of the control room

Activities dictated by the SAMG are coordinated by the TSC, which is located in emergency control centre under the administrative building. The tasks required in order to implement the strategy are carried out by managing operative staff from the CR or ECR. Local activities and necessary repairs are carried out in the corresponding rooms of the reactor chamber, machine room or outside buildings.

The CR and ECR are located in a clean part of the hall surrounding the reactor chamber. This part could be hit by radiation in case of a large-scale leakage of fission products from the containment. However, the CR and ECR are fitted with filtration venting systems.

The radiation situation in the CR and ECR after the containment bottom melts through has not been analyzed. The habitability or unhabitability of the CR and ECR has not been assessed using measurements of a radiation situation in case of such an adverse development of a severe accident. If the CR and ECR become inhabitable, the operational management team would have to be evacuated to the shelter, and any necessary actions (if permitted by current radiation) would be organized as response groups with defined measures of radiation protection.
In case both the CR and ECR need to be evacuated, the situation can be solved using design means. The TSC contains working stations that are configured to be a part of the information system; however, by changing the configuration of these working stations and moving operational management staff into the TSC, it is possible to implement SAMG strategy to a limited extent, using components of non-safety systems.

III.6.4.3 Conclusion on the adequacy of measures to restrict the radioactive releases

Although preventing a loss of containment integrity, as the last barrier preventing the leakage of fission products into the environment together with limiting the leakage are the main goals of the SAMG, the SAMG also describes strategies for stopping or reducing the leakage of fission products after a loss of containment integrity using all available means.

The opportunities to improve defence-in-depth in case of events that could lead to severe accident can be found in the following table. The table also contains areas that require the execution of additional analyses, which were not available at the time this assessment was executed.

Some measures (labeled in the note as "finding of the PSR") would be implemented even without this targeted evaluation, whose conclusions confirmed the effectivity and correctness of the previous decisions regarding the implementation of measures leading to increased strength of the original project.

Tab. 38: The opportunities to improve defence-in-depth in case of events that could lead to severe accident (to restrict the radioactive releases)

Improvement opportunity	**Corrective measure**	**Time period** **(Short-term I / Medium-term II)**	**Note**
Technical means	Hydrogen disposal system	II	Finding of the PSR ZKZ B462
Analysis	Localization of melt outside the RPV	II	Finding of the PSR Will be addressed in coordination with other operators of VVER1000

IV Conclusions

IV.1 General Conclusion

IV.1.1 Key provisions enhancing robustness (already implemented)

IV.1.1.1 Evaluation of resistance against earthquakes

There are no tectonic structures that could trigger strong earthquakes in the territory of the Czech Republic.

IV.1.1.1.1 Dukovany NPP site

There is a 95 % probability that an eventual earthquake will not exceed 6° MSIS-64 (PGAhor = 0.06 g) at the Dukovany NPP site. The actual resistance of systems, structures and components is higher, so there is a sufficient safety margin covering the remaining 5 % uncertainty. All safety important components and civil structures are currently being reinforced in all units of Dukovany NPP in order to be able to withstand the peak ground acceleration value of PGA = 0.1g (maximum design earthquake, MDE). More than 90% (including all technologies) of safety important components currently have got satisfactory documentation proving seismicity resistance. Modifications are being finalised and implemented for the remaining components. These modification will be completed by 2017 at all units.

IV.1.1.1.2 Temelín NPP site

Results from the detailed seismic zoning network also prove the overall seismic evaluation of Temelín site is correct. The units in Temelín NPP are equipped with a seismic monitoring system. There is a 95 % probability that any eventual earthquake will not exceed 6.5° MSIS-64 (PGAhor = 0.08 g) at the Temelín NPP site. The systems, structures and components necessary for performing safety functions are resistant at least up to 7° MSIS-64 (PGAhor = 0.1 g) so there is a sufficient safety margin covering the remaining 5 % uncertainty. Both historical data evaluation and long-term monitoring indicate that Temelín NPP site is seismically very calm.

IV.1.1.2 Evaluation of robustness against floods

IV.1.1.2.1 Dukovany NPP site

There is no threat of natural floods at the NPP Dukovany site. The NPP site is located on a plateau of altitude 383.5 - 389.1 m above sea level, while the main buildings in which safety important components are installed are situated at the upper edge of this interval. The nearest watercourse is the Jihlava river which is also used as a source of makeup service water for the power plant. Dalešice - Mohelno water reservoir system situated on the Jihlava river cannot affect the safety of Dukovany NPP in the event of extreme floods or in the event of break of both dams.

Dalešice dam, which is located upstream (approx. 4 km from the power plant, dam height 88 m) has a crest at an elevation of 384 m above sea level and its maximum water surface level (during overflow caused by floods) is located at an altitude of 381.5 m above sea level. Mohelno dam is located approx. 2 km downstream. Its crest lies at an elevation of 307.15 m above sea level and its maximum water surface level (during overflow) is at an altitude of 303.3 m above sea level, i.e. approx. 80 m lower than the Dukovany NPP buildings.

The annual long-term average precipitation can be characterized by the highest precipitation amount in summer, with the maximum in June (70 mm) and lowest total precipitation amount in winter with the minimum in January (21 mm). The drainage system network has been laid out in a branching pattern which provides gravity drain of rainwater from an area covering about 80 ha and it is connected to the main storm sewer system in front of the Dukovany NPP site.

The actual one-day storm rainfall total corresponds to a created water level of 77 mm on Dukovany NPP site (100 year total rainfall). The civil structures of Dukovany NPP were designed to withstand flooding up to the maximum water level of 115 mm (total precipitation per 24 hours in the case of a 10 000 year rainfall event). The difference between these values constitutes a sufficient safety margin. In addition, the corporate fire brigade has mobile equipment adjusted to drain local floodwater exceeding the 10 000 year rainfall maximum available on site.

IV.1.1.2.2 Temelín NPP site

There is no threat of natural floods at the NPP Temelín site. The NPP site is located on a plateau of altitude 507.3 m above sea level which is 135 m above the water level in Hněvkovice dam on the Vltava river. The safety of Temelín NPP was also evaluated with regards to eventual failure of the dams which are located upstream on the upper reaches of the Vltava river (Lipno on the Vltava river and Římov on the Malše river). In the event of a failure of Lipno dam the water level reached in Hněvkovice profile would correspond to a 10 000 year flood and it would cause flooding of a major part of the pumping station serving for the raw water supply to Temelín NPP. This would prevent standard raw water supply to Temelín NPP consequently leading to shutting both Temelín NPP units down. However, there is a sufficiently large volume of water stored on site which would allow cooling of both reactors to cold shutdown status

Flooding of safety important buildings from the gravity sewage system is not possible even in the event of extreme precipitation if regularly maintained. The water drainage system at Temelín NPP is cascaded having buildings which are nuclear safety most vital placed at the highest elevation and sloping down to the borders of the site. This enables to keep natural gravity based drainage even in the event of stormwater drainage system failure. The civil structures on Temelín NPP site are designed to withstand flooding in the event of one-day maximum rainfall at which the maximum level of 88.1 mm of 10,000-year precipitation is reached even if the stormwater drainage system is out of order. In addition, the corporate fire brigade has mobile equipment adjusted to drain local floodwater exceeding the 10 000 year rainfall maximum available on site.

IV.1.1.3 Evaluation of robustness against extreme weather conditions

IV.1.1.3.1 Dukovany NPP site

Weather phenomena parameters for Dukovany NPP site are based on statistical processing of annual extremes of relevant meteorological parameters measured on Dukovany NPP site or at meteorological stations in the surrounding region over a period of at least 30 years. The weather effect design load is based on a 100-year average recurrence interval. The extreme design load is calculated on the basis of 10 000-year average recurrence interval. Buildings of seismic category 1 must withstand the effects of extreme design load in such a way that the function of nuclear safety important systems is not compromised. Other buildings are loaded at the design level.

The particular values of derived weather condition extremes on the NPP Dukovany site, including the corresponding values of design load and extreme design load of the buildings, are stated in part II.4.1.1. of this report.

Some partial deviations between the actual robustness of selected buildings and the required values of robustness under extreme load (which, however, cannot influence the performance of safety functions) are being solved in the current project of supplemental seismic reinforcement of safety important components and civil structures, which will also increase resistance to extreme weather conditions.

IV.1.1.3.2 Temelín NPP site

The weather phenomena load is based on statistical processing of data series over at least 30 years of measuring such events on Temelín NPP site or in a location with a similar character of landscape.

The particular values of derived weather condition extremes on Temelín NPP site, including the corresponding values of design load and extreme design load of the buildings, are stated in part III.4.1.1. of this report.

Actual resistance of seismic category 1 buildings is higher than the values calculated for extreme design loads resulting from extreme weather conditions.

IV.1.1.4 Evaluation of robustness against loss of electrical power supply

IV.1.1.4.1 Dukovany NPP

The power supply sources in Dukovany NPP provide a sufficient design robustness and safety assurance level in the event of a loss of off-site power. They have been designed with a high level of independence, mutual backup and redundancy; see operating and backup sources for their house consumption, as well as emergency sources of alternating and direct current, and the secured power supply systems, which are used to power the safety important systems and components.

There is a higher design robustness against loss of off-site power during power operation of the unit (additional defence in depth barriers), than during refuelling outages. The least favourable scenario in terms of safety assurance is a loss of off-site power for all units at the same time.

There is in totality 12 emergency sources of alternating current supply (DGs). If a loss of off-site power mode occurs, Dukovany NPP units can be maintained in safe operational mode on a long-term basis or cooled down to a cold shutdown state or safely maintained in an outage mode (with power supply to all necessary mechanical systems and I&C systems ensured) on condition that at least one of these DGs is started up in each unit. Each DG has a sufficient reserve of diesel fuel that could last for 6 to 7 days without the need for external refuelling.

Emergency sources of uninterrupted direct current (accumulator batteries) are available as a power supply to safety systems and other safety related systems in the event of station blackout (SBO). The accumulator batteries cannot be recharged without the correspondent DG in operation and they can therefore last for hours or tens of hours depending on the actual load. Such a period of time is sufficient for the renewal of power supply to Dukovany NPP units from the nearby Dalešice or Vranov hydroelectric power plants.

IV.1.1.4.2 Temelín NPP

The power supply sources in Temelín NPP provide a sufficient design robustness and safety assurance level in the event of a loss of off-site power. They have been designed with a high level of mutual independence of working and backup sources to cover their house consumption, and also with redundancy of secured power supply systems which are used to supply power to the safety important systems and components. They dispose of their own emergency power sources (DGs and accumulator batteries). Each unit uses a power supply covering its house consumption which prevents propagation of electrical faults inside Temelín NPP.

There is a higher design robustness against loss of off-site power during power operation of the unit (additional defence in depth barriers), than during refuelling outages. The least favourable condition in terms of safety assurance is a loss of off-site power for all units at the same time.

There is in totality 8 emergency alternating current sources available on site (3 safety DGs per unit and 2 common DGs for both units). If a loss of off-site power mode occurs, Temelín NPP units can be maintained in a safe operational mode on a long-term basis or cooled down until a cold shutdown state or safely maintained in an outage mode (with power supply to all necessary mechanical systems and I&C systems ensured) on condition that at least one of these DGs is started up in each unit. Each of these DGs has sufficient reserve of diesel fuel that would last for 2 to 3 days without the need for external refuelling. There are additional diesel fuel reservoirs for further extension of operation of the DGs available on site.

Accumulator batteries providing electricity supply to safety systems can last several hours without recharging, depending on the load. The accumulator battery runtime can be increased significantly by controlled relieving of the battery load, gradual use of individual divisions and utilization of accumulator batteries of safety related systems with greater capacity. Such a period of time is sufficient for the renewal of power supply to Temelín NPP units from the nearby Lipno hydroelectric power plant.

As an alternative, it is possible to use other alternating current sources available on Temelín NPP for long-term recharging of the accumulator batteries. This has been proposed as a measure for increasing the NPP robustness against the loss of off-site power.

IV.1.1.5 Evaluation of robustness against loss of ultimate heat sink

IV.1.1.5.1 Dukovany NPP

Ultimate heat sink for reactors at Dukovany NPP is provided by ambient atmosphere. Unused heat during power operation of the unit, or decay heat after reactor shutdown as the case may be, can be transferred to the ultimate heat sink – atmosphere – in several ways. Heat transfer between heat sources and atmosphere is ensured by the ESW (essential service water) system.

The on-site water reserve is sufficient for about 39 days of operation of the ESW system, ensuring removal of decay heat from shutdown reactors of Dukovany NPP without external water replenishment to the ESW system. A total of 12 ESW pumps are available for each main power generation unit (2 reactors). Loss of all ESW pumps could be caused by concurrent loss of electrical power to both units of the given main power generation unit.

The robustness of Dukovany NPP against eventual loss of all ESWs corresponds to the SBO occurrence scenario. If the loss of ESW system was not combined with SBO, an alternative method of heat accumulation from spent fuel storage pools (SFSP) to emergency core cooling system (ECCS) accumulator tanks, or eventual replenishment of coolant evaporated from the SNF storage pools from bubbler trays can be used. The accumulation capability with full emergency core cooling system accumulator tanks covers more than 4 days, the coolant reserves in bubbler trays for the replenishment of evaporated coolant can last for approximately 13 days. As an alternative, it is possible to use firefighting equipment to replenish the evaporated coolant and maintain the spent fuel temperature in the spent fuel storage pool.

IV.1.1.5.2 Temelín NPP

The ultimate heat sink for reactors at Temelín NPP is provided by ambient atmosphere. Unused heat during power operation of the unit, or decay heat after reactor shutdown as the case may be, can be transferred to the ultimate heat sink – atmosphere – in several ways. Heat transfer between safety important heat sources and atmosphere is ensured by the ESW system through spray cooling water basin (CBSS).

The on-site water reserve in CBSS in Temelín NPP is sufficient for about 30 days of operation of the ESW system, ensuring removal of decay heat from shutdown reactors without external water replenishment to the ESW system. A total of 6 ESW pumps are available for each unit. Considering the spatial separation of systems and pumps and the independence of the electrical power supply and other support systems, concurrent unavailability of all ESW pumps is extremely unlikely. The fundamental safety functions can be fulfilled even if there is only one pump in operation in one division of the ESW system.

IV.1.1.6 Evaluation of severe accident management measures

IV.1.1.6.1 Dukovany NPP and Temelín NPP

Both NPPs have implemented practically identical 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 in both NPPs is ensured by means of a robust set of measures covering personnel, administrative and technical measures.

As far as the personnel is concerned, these measures include the existence of an emergency response organization and ensuring of activities related to the individual functions; the administrative measures are represented by the implementation of relevant procedures, guidelines and instructions using the capacities of technical support centres and the technical measures include assurance of functionality of the required scope of technical means for the implementation of strategies. Performance of interventions upon the occurrence of an emergency event is ensured during the first (preventive) stage of the evolvement of the event by the continuous shift operation personnel. In the case that the extent of the event develops beyond the scope of the capacity of the continuous shift personnel, stage two (mitigation of consequences) commences and emergency response organization is activated. In such a case, the responsibility for the management of interventions is taken over by the emergency staff with the assistance of the technical support centre.

In the case of an emergency event, all the necessary activities would be managed and performed from secured locations. The technical support centre and emergency staff who are responsible for the management of strategies according to SAMG are based in the emergency control centre which is a secured workplace habitable even in the event of the release of radioactive substances into the atmosphere. Remote activities for the implementation of strategies would be performed by the shift personnel from the Main Control Room (MCR) or Emergency Control Room (ECR). A project enabling occupancy of these control centres is being finalized at Dukovany NPP. Local activities and possible equipment repairs in the corresponding areas of the reactor hall, generator room or external buildings would be carried out by emergency response teams based in the technical support centre.

The accident management concept of Dukovany NPP and Temelín NPP is based on a symptomatic approach. Strategies for the management of technological accidents have been developed and included in EOPs. Their highest priority is to restore heat removal from the active zone of the reactor and to prevent any possible damage to the first barrier against the release of fission products (fuel cladding). Strategies for the mitigation of consequences of severe accidents have been developed and included in SAMG. Their highest priority is to prevent any possible damage to the third barrier preventing the release of fission products (containment) which is the last barrier remaining intact at the given time. EOPs and SAMG are updated on a regular basis to incorporate both findings obtained from the practicing of their use on a simulator or during emergency drills, as the case may be, and external knowledge.

If the safety of a unit or site is compromised or if a situation which the shift personnel cannot manage occurs, the emergency preparedness system is implemented. When one of the emergency event levels (Alert, Site emergency, General emergency) has been declared, the emergency response organization (ERO) is activated. The internal part of this organization (IOER) consists of shift personnel and its emergency part (SOER) consists of NPP technical staff specialists.

A system of qualification requirements has been implemented to select shift personnel and experts for SOER and additional criteria taking their knowledge and expertise into account are also considered. The preparedness of shift and technical personnel to manage technological accidents is regularly tested during full-scope simulator training with the participation of the Technical Support Center staff and during emergency drills.

The organisation of emergency event management (including severe accidents) is defined in the Internal emergency plan approved by SÚJB (State Office for Nuclear Safety) in both NPPs.

When emergency conditions occur (design basis or beyond design basis accidents without damage to fuel rods), all currently available technical equipment is employed within the scope of its design specification in order to meet the EOP requirements. SAMG expect that the required actions will be performed using all available systems and components, or better, all available technical equipment even outside design specifications.

There is a corporate fire brigade unit available at both NPP sites which operates appropriate fire fighting equipment at their disposal. The fire brigade personnel is trained to intervene at any part of the sites. The pumping equipment of the fire brigade is among the main mobile non-technological equipment available for transportation and pumping of various substances.

The accident management program in Dukovany NPP and Temelín NPP has been analytically supported. Analytical support is based on probabilistic and deterministic approach based on the selection of the most likely emergency scenarios leading to severe accidents which are subsequently analysed in deterministic way using integral computation codes. The analytical support results to a summary of findings based on the understanding of phenomena and their timing during severe accidents, identification of possible project weaknesses, specification of activities to mitigate consequences of severe accidents, validation of severe accident response activities and determination of the source term for the evaluation of possible radiological consequences. A simulation tool to display phenomena during particular severe accident scenarios is also available.

IV.1.1.7 VVER 440/213 reactor containment specification

VVER 440/213 reactors which are in operation in Dukovany NPP are characterised by their specific construction of the protective envelope – containment equipped with the “bubbler system”. It is a passive condensation system whose basic function is to reduce pressure of the air – water vapour mixture in the hermetic reactor zone following a maximum design basis accident (guillotine rupture of the primary piping with the diameter of 500 mm) by condensing water vapour in special trays filled with boric acid (H_3BO_3) solution with subsequent isolation of the uncondensed gasses in hermetic air traps with check valves. By creating an underpressure against the surrounding atmosphere the system concurrently minimizes possible radioactivity releases outside the hermetic space. The system has been designed to maintain its integrity in pressure and thermal conditions which occur in the hermetic zone after a maximum design basis accident.

The thermodynamic principle on which the bubbler-condenser system is based is identical to the function of pressure suppression containment used in the western boiling water reactors. With regards to limited data concerning the experimental verification of the system from the original designers and to the need to extend the knowledge and ability to model the integral behavior of the system and partial physical phenomena in the conditions of large and small LOCA, a range of international projects and studies were organized in the 90s. These were participated in by countries operating these reactor types and renowned western institutions, such as SIEMENS/KWU, EdF Empresarios Agrupados, GRS, IRSN, etc. The first series of studies was carried out within the framework of the Extrabudgetary Programme, organized by the International Atomic Energy Agency:

- Ranking of Safety Issues for VVER 440 Model 213 Nuclear Power Plants IAEA-Report VVER-SC-108 1995-02-21
- Strength Analysis of the Bubbler Condenser Structure of VVER-440 Model 213 Nuclear Power Plants, IAEA-TECDOC-803, Vienna 1995
- Report of a Consultants' Meeting on the Review of Bubbler Condenser Structure Integrity Calculations, IAEA/ TA-2485 TC Project RER/9/035 12-16 June 1995.

A fundamental experimental verification of the performance and structural and strength characteristics of the bubbler-condenser system was carried out within PHARE and TACIS programmes of the European Commission. An experimental stand was built in Elektrogorsk EREC research center in the Russian Federation within the framework of PHARE/TACIS 2.13/95 "Bubbler Condenser Experimental Qualification – BCEQ" project with the aim of verifying integral behavior of the bubbler-condenser system and obtaining qualified experimental data suitable for the validation of calculation programmes. This experimental facility together with smaller models of the parts of the bubbler-condenser system in VUEZ Tlmače (Slovakia) and SVUSS Běchovice (CR) experimental facilities allowed:

- experiments simulating integral thermohydraulic and hydrodynamic interaction of water vapor with the bubbler tower construction (managed by Siemens/KWU)
- implementation of static tests for the verification of strength characteristics of the bubbler-condenser system construction in VUEZ Tlmače (managed by: Empresarios Agrupados).
- small-scale experiments to study partial thermohydraulic phenomena and verification of instrumentation in SVUSS Běchovice (managed by: Electricité de France).

Experimental work within the framework of the project was concurrently accompanied by analytical studies in which VEIKI research institute, Budapest (Hungary) also participated. The project results were summed up in the series of research reports, among others:

- Final Project Report, K.Kühlwein *et al.*,/BC-D-SI-EC-0535/, December 1999
- Final Thermal-hydraulic Test Report, D.Osokin *et al.*, /BC-D-SI-EC-0028/, November 1999
- Parallel Thermal-Hydraulic Test Analyses, M.Suchanek *et al.*, /BC-D-SV-EF-0011/
- Experimental Qualification of Measurement Techniques (visualization, strain
- gauges) I.Batalik *et al.*
- Small Scale Test Final Report; J.Batalik, J.Murani *et al.*December 1999, /BC-D-EA-EC-0015/
- Static Structural Tests Final Report, Rev.1, December 1999.

The project provided unambiguous, factual and objective proof that the bubbler-condenser system is qualified for the conditions of maximum design basis accident of VVER 440/213 reactors, and that it is capable of carrying out its safety function – maintaining integrity of the hermetic reactor zone and prevention of release of radioactive substances into the environment – under these conditions.

The BCEQ project was not the last project to be organised to verify the bubbler-condenser system performance. The “Answers to Remaining Questions on Bubbler-Condenser” project was commenced in the years 2001-2002 under the sponsorship of CSNI OECD/NEA international committee. Its aim was to answer some additional questions regarding the conservativeness of the results of the BCEQ project, experiment criteria, nonhomogeneities in the stream and temperature field in the bubbler tower volume and particularly verification of the performance of bubbler condenser system in the conditions of long-lasting small LOCA.

The project was initiated by the supervisory bodies of the CR, SR and Hungary and financed by power generation companies in these countries. The project steering group consisted of one representative of each of the abovementioned supervisory bodies, as well as experts from German GRS, French IRSN, US DOE and the EU. The project also included three additional experiments in EREC experimental stand, namely:

- steam pipe rupture
- medium LOCA (rupture of a pipe 200 mm in diameter)
- small LOCA (rupture of a pipe 90 mm in diameter).

The conclusions of these tests and the final stance of the Steering committee to the given questions are summed up in a report called “*Answers to Remaining Questions on Bubbler-Condenser”, Activity Report of the OECD NEA Bubbler-Condenser Steering Group, NEA/CSNI/R(2003)12, January 2003***.** The project steering committee concluded the project by stating in the report that additional experiments proved that the load which the bubbler-condenser system is exposed to under design basis accidents conditions does not pose a threat to the integrity of the bubbler-condenser system. **This project conclusion was also accepted by OECD/NEA CSNI committee.**

The findings of stress tests prove that, besides its classic safety function, the bubbler-condenser system may play an important role in the conditions of design basis accidents even in the event of a severe accident. It is due to its volume, amount of water with suspensory boric acid concentration and the surface of internal building and technological constructions which significantly influence the critical concentration risk, hydrogen explosion and eventual release of radioactive substances outside the hermetic reactor zone**.**

IV.1.2 Safety issues

The stress test assessment results proved the existence of safety and time margins and the high robustness of both NPPs against extreme external influences. The previously adopted decisions to implement measures leading to the increase of robustness of the original NPP designs were proved correct, especially in connection with seismic risk. No issue requiring immediate intervention has been found in the NPPs. Both NPPs are able to deal with even highly improbable extreme accidents safely without endangering their surroundings. Despite the above-mentioned statements, the stress tests identified opportunities for further enhancement of the robustness of both NPPs against extreme external influences.

IV.1.3 Possible safety improvements and further work forecasted

Opportunities for improvements are of both organizational and technical character. These potential measures will be subject to further analyses as regards their efficiency. Technical measures concerning modifications to the current NPP design will be subject of feasibility studies including a proposal of specific design modifications which is to be approved before implementation by SÚJB.

More detailed explanations of the proposed modifications are included in the relevant parts of the report, including their classification as short-term or long-term with regards to their implementation. Measures which are being implemented are the result of a periodic safety review.

IV.1.3.1.1 Possible safety improvements in the earthquake field

Dukovany NPP:

- Seismic robustness of critical systems, civil structures and components
- Seismic robustness of non-seismic equipment
- Process operating instructions concerning earthquakes
- EDMG guidelines for the use of alternative means
- Ability of emergency response organization to operate outside the emergency control centre
- Seismicity robustness of corporate fire brigade unit building
- Alternative means of communication after a seismic event
- Analysis of danger to shelters during a seismic event
- Ensuring availability of a sufficient number of personnel after a seismic event
- Access to buildings, availability of heavy machinery

Temelín NPP:

- Alternative replenishment of diesel fuel from a tank for a long-term operation of DG
- EDMG guidelines for the use of alternative means
- Capability of emergency response organization functioning outside the emergency control center
- Seismic robustness of corporate fire brigade unit building
- Alternative means of communication after a seismic event
- Analysis of shelters during a seismic event
- Ensuring a sufficient number of personnel after a seismic event
- Access to buildings, availability of heavy machinery

IV.1.3.1.2 Possible safety improvements in the flood field

Dukovany NPP:

- Filling of SFSPs by gravity drainage from bubbler trays
- Removal of heat from the coolant in the SFSPs by means of coolant replenishment and its accumulation in TH tanks
- EDMG guidelines for the use of alternative means
- Analysis of vulnerability of shelters during floods

Temelín NPP:

- Improvement of robustness of DG building against external floods
- Ability of emergency response organization to operate outside the emergency control center
- EDMG guidelines for the use of alternative means
- Analysis of vulnerability of shelters during floods

IV.1.3.1.3 Possible safety improvements in the extreme weather conditions field

Dukovany NPP:

- Implement measures for the diversion means of the UHS (to the cooling tower)
- Process operating instructions for extreme events (wind, temperature, snow)
- EDMG guidelines for the use of alternative means
- Ensuring a sufficient number of personnel after extreme events
- Robustness of buildings (corporate fire brigade unit, central pumping station, main power generation unit, etc.) against extreme conditions
- Elaboration of methodologies for the assessment of external influences, verification of performed analyses, eventual technical measures

Temelín NPP:

- Alternative replenishment of diesel fuel from a tank for a long-term operation of DG
- Ensuring of personnel availability during extreme events
- Elaboration of methodologies for the assessment of external influences, verification of performed analyses, eventual technical measures

IV.1.3.1.4 Possible safety improvements in the case of loss of off-site power or the ultimate heat sink

Dukovany NPP:

- Ensure sufficient power supply source for secured power supply systems of the first category and selected secured power supply appliances of the second category.
- Ensure sufficient source for steam generator refilling
- Analyze the possibility of alternative refilling of the reactor using a pump and a new pipe route
- Implement diverse (to the cooling tower) UHS means
- Develop a procedure for restoring power supply to all units after a SBO
- Develop a procedure for the loss of UHS and ESW systems in all 4 units
- Develop a procedure for the refilling of steam generators using fire fighting equipment
- Filling an open reactor and SFSP by gravity drainage from bubbler trays
- Removal of heat from the coolant in the SFSPs by means of coolant replenishment and its accumulation in TH tanks
- EDMG guidelines for the use of alternative means
- Analysis of battery lifetime upon the use of controlled relief of its load, addition of operating instructions, change of emergency lighting connection and operation
- Ensure alternative power supply to shelters and telephone switchboards
- Ensure alternative source of power supply for the technical system of physical protection
- Ensure sufficient number of personnel during long-lasting SBO

- Capability of emergency response organization functioning outside the emergency control center

Temelín NPP:

- Implement measures for the diversion means of the UHS (to the cooling tower)
- Develop a procedure for the loss of UHS and ESW systems in all 4 units
- EDMG guidelines for the use of alternative means
- Alternative replenishment of water to steam generator/SFSP/primary circuit (with unsealed primary circuit)
- Alternative source for the recharging of accumulator batteries and powering selected appliances
- Alternative replenishment of diesel fuel from a tank for long-term operation of DG
- Analysis of heat removal from I&C systems following a loss of ESW
- Reconnection of containment insulation fittings of air-conditioning systems to accumulator batteries
- Use of safety DGs of the neighboring unit during SBO
- Analysis of battery lifetime upon the use of controlled relief of its load, addition of operating instructions
- Containment insulation procedure in shutdown state
- Removal of heat from the SFSP without refilling
- Procedure for restoring power supply after a SBO at all units
- Ensuring sufficient number of personnel during long-lasting SBO
- Analyses of options of shift personnel during SBO at all units
- EDMG guidelines for the use of alternative means
- Alternative sources and means of communication after a seismic event
- Work out a procedure for unit operation during long-term power supply from emergency sources

IV.1.3.1.5 Possible safety improvements in the severe accidents management field

Dukovany NPP:

- Ensuring of MCR habitability
- Oxygen regeneration in shelters
- Increase of the hydrogen removal system capacity
- Cooling of melt from outside the reactor pressure vessel
- Supplementation of measurements concerning radioactive situation and stated of the SFSP
- Draw up “shutdown SAMG” for shutdown / SA in SFSPs
- Ensure alternative means of warning and informing Dukovany NPP personnel and inhabitants in the emergency planning zone.
- EDMG guidelines for the use of alternative means
- Build an emergency control center outside Dukovany NPP site.
- Improve training and practice of technical support center in the SA field
- Draw up agreements with external bodies (integrated rescue system, army) and nearby NPPs.

Temelín NPP:

- Alternative replenishment of water into the containment sump
- Implementation of hydrogen removal system in the containment for SA
- Localization of melt outside the reactor pressure vessel

- Verification of equipment functioning in beyond design operating conditions
- Analysis of radiation situation at MCR/ECR during a SA
- Process “shutdown SAMG” (damage to fuel during an open reactor / in the SFSP)
- EDMG guidelines for the use of alternative means
- Staffing of the emergency response organization by qualified and well-trained personnel
- Capability of emergency response organization functioning outside the emergency control center
- Draw up agreements with external bodies (integrated rescue system, army) and nearby NPPs.

Besides the above-mentioned measures proposed at both NPPs, SONS is considering to suggest to Temelín NPP to analyse the possibility and various alternatives of modifications to complete the original containment design with the feasible venting option for the case of severe accidents (Type II). This modification has already been introduced in various modification and upgrades in a range of western NPPs. The procedure should be coordinated with other VVER-1000 type NPP operators and regulators.

SÚJB will also suggest to ČEZ a.s. that they should consider the establishment of a common VVER operator center for mutual aid in the case of severe accidents (Dukovany, Bohunice, Mochovce, Paks) with regards to the short geographical distance between them. The common center for severe accidents would allow to solve in effective way the need for the purchase of mobile diesel generators and other financially demanding equipment which would probably never be used in standard situations.

IV.2Conclusion

The evaluation of safety margins of Dukovany NPP and Temelín NPP under extreme weather conditions, and with loss of off-site power or loss of the ultimate heat sink, and of their ability to handle situations when the scenario develops into a severe accident, confirmed the presence of safety margins and the sufficient robustness of barriers ensuring defence in depth in the areas of personnel, administrative and technical assurance of accident handling in the majority of accident scenarios.

It can be said that, despite the robustness of the barriers, opportunities for further increases in safety for highly improbable, beyond design basis events have been identified, based on the results of the evaluation of safety margins for initiating events, loss of safety functions and measures for the handling of beyond design basis accidents and severe accidents of Dukovany NPP and Temelín NPP.

The importance of each identified potential was defined in terms of the size of safety margins, i.e. resistance against possible loss of ability to provide elementary safety functions and readiness to deal with the situation that has occurred. When assessing the significance of individual risks, the number of defences in depth which would have to fail to allow the occurrence of such a situation and the period of time for which the unit is able to withstand the situation with the existing safety margins were taken into consideration. During this period of time, it is necessary to have sufficient means available to provide the required functions or to adopt subsequent safety measures in order to mitigate radiation exposure and protect the people.

In the majority of accident scenarios, external risk evaluation and analysis of safety margins against these risks proved that the current design of both power plants provides sufficient margins in parameters and time needed for the personnel to react in order to avoid severe accidents. Strengths of both power plants from an external risk point of view include:

- robustness and conservativeness of the design able to withstand extreme conditions
- design which undergoes continuous inspection and verification with up-to-date safety requirements
- continual process of including new safety requirements
- sites with minimum seismic risk (the robustness of Dukovany NPP is currently being increased to PGA 0.1 g)
- sites in which the possibility of external floods can be practically excluded
- two large raw water reservoirs for both power plants
- large reserves of cooling water inside the power plants
- compact SFSPsensuring subcritical state of the fuel even if ordinary water is used
- Dukovany NPP has a large volume of hermetic space (bubbler condenser system) and a smaller source term (lower power output of the reactor).
- SFSPin Temelín NPP is located in a fully pressured containment, etc.

It must be highlighted in the logic and the aims of the evaluation of the Czech NPP stress test, that the territory of the Czech Republic is not subject to extreme natural phenomena such as earthquake, flood, etc. which could compromise safety of the NPPs. This naturally applies to Dukovany NPP and Temelín NPP sites, the selection of which respected the official criteria for NPP siting in terms of nuclear safety, which has been used in the Czech Republic in connection with IAEA recommendations since the 1970s.

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