Czech Republic

Extraordinary National Report

under the

Convention on Nuclear Safety

February 2012

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INTRODUCTION

This report is the Extraordinary National Report of the Czech Republic elaborated for the purposes of an extraordinary review meeting of the contracting parties to the Convention on Nuclear Safety to be held in August 2012. The objective of the report is to describe the level of nuclear safety of operated nuclear power plants from the viewpoint of their resistance against selected extreme phenomena in the Czech Republic, as at 29 February 2012. This extraordinary review process has been initiated by the events in Japan following the destructive earthquake on 11 March 2011.

The structure of this National Report is based on recommendations made by the International Atomic Energy Agency (IAEA) and it consists of 6 chapters: 1/External events, 2/Design basis, 3/ Management of severe accidents and recovery of safety functions of units on the site, 4/ National organization, 5/ Emergency preparedness and emergency response and 6/ International cooperation.

Similarly, the report observes IAEA recommendations about the structure of the individual chapters. Each chapter starts with a general introduction, which in the chapters 1, 2, 3 and 5 also describes relevant criteria established by the national legislation, particularly by the Atomic act and its implementing regulations. The part 2 of each chapter is elaborated by the nuclear power plant Licensee; the part 3 is elaborated by the state regulatory authority – the State Office for Nuclear Safety, including the evaluation of the part elaborated by the Licensee.

Each chapter is concluded, again in agreement with the IAEA recommendations, with a summary which covers the individual described activities and their current status. The topics of the 6 chapters are related to the National Report of the Czech Republic developed in April 2010 for the 5th review meeting held in April 2011.

There are two nuclear power installations in operation in the Czech Republic that are subject to the Convention on Nuclear Safety; they are both operated by ČEZ, a. s. - the Licensee authorized to operate nuclear installations in agreement with the wording of the Atomic act.

Specifically, the nuclear installations are:

- the Dukovany Nuclear Power Plant (Dukovany NPP) with four units with the VVER 440/213 reactors

The units were put into permanent operation as indicated below (the year in the brackets indicates the year of issuance of the certificate of practical completion):

Unit 1 - 1985 (1988)

Unit 2 - 1986 (1988)

Unit 3 - 1987 (1989)

Unit 4 - 1987 (1990)

and

- the Temelín Nuclear Power Plant (Temelín NPP) with two production units with the VVER 1000/320 reactors. Both units were put into permanent operation in 2004. The certificates of practical completions for both units were issued in 2006.

LIST OF ABBREVIATIONS

AAC	Additional source of alternative current
AC	Alternating Current
ADR	Agreement about transport of dangerous consignments “Accord Dangerous Route“
AIRS	Advanced Incident Reporting system database
AQG	Atomic Question Group
ASSET	Assessment of Safety Significant Events Team
Atomic act	Act No. 18/1997 Coll. , on peaceful utilization of nuclear energy and ionizing radiation (Atomic act), as amended
BCEQ	Bubble Condenser Experimental Qualification
BDBA	beyond-design basis accident
BWR	Boiling Water Reactor
CB	Crisis Board
CBSS	Cooling Basins with Sprinkler System
CNRA	Committee on Nuclear Regulatory Activities
CPS	Central Pumping Station
CR	Czech Republic
CRPPH	Committee on Radiation Protection and Public Health
CSNI	Committee on the Safety of Nuclear Installations
CT	Cooling Tower
CTG	Czech transmission grid
CTMT	Containment
CVŘ	Research Centre Řež s.r.o
ČEZ, a. s.	trade name of the utility joint-stock company ČEZ, a. s.
ČSKAE	Czechoslovak Commission for Atomic Energy
ČSN	Czech technical standard
ČVUT	Czech Technical University
DBE	Design Basis Earthquake
DE	Design Earthquake
DG	Diesel generator
DGS	Diesel generator station
DSR	Detailed Seismic Regionalization
EC	European Commission
ECC	Emergency Control Centre
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
EDMG	Extensive Damage Mitigation Guideline
EDU	Dukovany NPP
EE	Extraordinary Event
EFWP	Emergency Feed Water Pump
EGP	Energoprojekt Praha
ELI	Hydropower station Lipno
EN	European standard
ENC	European Nuclear Council – association of CEOs of NPP operators in Europe
ENIQ	European Network for Inspection Qualification
ENISS	European Nuclear Installations Safety Standards - association of operators for harmonization of European nuclear safety standard
ENS + ČNS	European and Czech Nuclear Society
ENSREG	European Nuclear Safety Regulators Group
E.ON	Power company
EOP	Emergency Operation Procedure
EP	Emergency Preparedness
EPRI	US Electric Power Research Institute
EPS	Emergency Power Supply
EPZ	Emergency Planning Zone
ERB	Emergency Response Board
EREC	Elektrogorsk Research & Engineering Center

ESW	Essential Service Water
ETE	Temelín NPP
EU	European Union
EUR	European Utility Requirements – association of West European operators seeking to standardize safety requirements for new generation of nuclear reactors
EURATOM	European Atomic Energy Community
Eurelectric	association of the electricity industry in Europe
FORATOM	European Atomic Forum - nuclear industry organization
FRS	Fire Rescue Service
FWT	Feedwater Tank
GRS	Geselschaft für Anlagen- und Reactorsicherheit
HA	Hydro Accumulator
HC	House Consumption
I&C	Instrumentation and Control system
I.C.	Primary circuit
IAEA	International Atomic Energy Agency
IEC	International Electrotechnical Commission
II.C.	Secondary circuit
INES	International Nuclear Event Scale
INPO	Institut of Nuclear Power Operators
INSAG	International Nuclear Safety Advisory Group
INSC	Instrument for Nuclear Safety Cooperation
IOER	Internal Organization of Emergency Response
IPERS	International Peer Review Service
IPPAS	International Physical Protection Advisory Service
IRRS	IAEA Integrated Regulatory Review Service
IRRT	International Regulatory Review Team
IRS	Incident Reporting System
IRS	Integrated Rescue System
IRSN	from the French "L'Institut de Radioprotection et de Sûreté Nucléaire"
ISO	International Standard Organization
KRUC	Seismic station in CR
LBB	Leak Before Break
LFRS	Local Fire Rescue Service
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LTO	Long Term Operation
MCP	Main Circulating Pump
MCR	Main Control Room
MDE	Maximum Design Earthquake
MPU	Main Production Unit
MSK-64	Medvedev Sponheuer Karnik (scale of seismic intensity)
NEWS	Nuclear Events Web-based System
NPP	Nuclear Power Plant
NPT	Nuclear Non-Proliferation Treaty
NS	Nuclear Safety
NSRS	Non-safety Related Systems
NucNet	International Communications Network for Nuclear Energy and Ionising Radiation
NUMEX	Nuclear Maintenance Experience Exchange - international association of operators dealing the maintenance of nuclear installations
NUSSC	Nuclear Safety Standards Committee
ODM	Operational Decision Making
OECD-NEA	Organisation for Economic Co-operation and Development – Nuclear Energy Agency
OPIC	Operating and Information Center
OSART	Operational Safety Review Team
OSMIR	Operational Safety Review Team Mission Results
PAMS	Post-Accident Monitoring System
PERIZ	Periodic Integral Tightness Test
PGA	Peak Ground Acceleration

PHARE	Programme of technical support organized by the European Commission
PORV	Power Operated Relief Valve
PRZR SV	Pressurizer Safety Valve
PSA	Probabilistic Safety Assessment
PSJ	Pumping Station Jihlava
PSR	Periodic Safety Review
PSV	Pressurizer Safety Valve
PWR	Pressurized Water Reactor
Ra	Radioactive
RASSC	Radiation Safety Steering Committee
RC	Reactor Core
RHWG	Reactor Harmonization Working Group
RMN	Radiation Monitoring Network
RPV	Reactor Pressure Vessel
RTS	Reactor Trip System
SAGTAC	IAEA's Standing Advisory Group on Technical Assistance and Cooperation
SALTO	IAEA's Extrabudgetary Programme on Safety Aspects of Long Term Operation
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout (Total Loss of House Consumption)
SBSA	Steam Bypass Station to Atmosphere
SDEOP	Shutdown EOP
SE	Shift Engineer
SEFWP	Super Emergency Feedwater Pump
SFSP	Spent Fuel Storage Pool
SG	Steam Generator
SG	Safety Guide
SL2	Maximum Design Earthquake
SMS	Seismic Monitoring System
SNF	Spent Nuclear Fuel
SOER	Standby Organization of Emergency Response
SPS	Secured Power Supply
SPSS	Secured Power Supply Systems
SR	Slovak Republic
SSC	System, Structure, Component
SÚJB	State Office for Nuclear Safety
SÚJCHBO	National Institute for Nuclear, Chemical and Biological Protection
SÚRO	National Radiation Protection Institute
SW	Software
TACIS	Technical Assistance to Commonwealth of Independent States
TECDOCs	Technical Documents of the IAEA
TG	Turbo Generator
TH	Low-pressure part of ECCS
TRANSSC	Safe Transport of Radioactive Materials Steering Committee
TSC	Technical Support Center
TSPP	Technical System of Physical Protection
UHS	Ultimate Heat Sink
ÚJV Řež a. s.	Nuclear Research Institute in Řež
UPS	Uninterruptible Power Supply
US DOE	United States Department of Energy
US NRC	United States Nuclear Regulatory Commission
USS	Unit Shift Supervisor
VBC	Vacuum-bubbler condenser
VGB	German scientific-technical organization , dealing also with nuclear power industry
VŠB	VŠB-Technical University of Ostrava
VVER (or WWER)	type identification of Water-Water Energetic Reactors designed in the former Soviet Union
WANO	World Association of Nuclear Operator
WANO SOER	Significant Operating Experience Report of WANO
WASSC	Safety of Radioactive Waste Steering Committee
WENRA	Western Nuclear Regulatory Association

WGWD	Working Group on Waste and Decommissioning
WIG	WENRA Inspection Group
WNA	World Nuclear Association – world´s organization of the nuclear industry
WPR	WANO Peer Review
XL	Bubbler Condenser System

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1. EXTERNAL EVENTS

1.1 INTRODUCTION

Evaluations of nuclear safety for the purposes of the licensing procedure for constructions with nuclear installations have a long tradition in the Czech Republic. As early as in the beginning of 1970s the building act of the former Czechoslovakia established the obligation to submit for the site approval procedure, construction permit procedure and for the commercial operation license three types of safety reports: Pre-siting Safety Report, Preliminary Safety Analysis Report and Final Safety Analysis Report. The first generally binding regulations were issued in 1970s for the purposes of their evaluation by the state regulatory body – the Czechoslovak Commission of Atomic Energy, including:

- ČSKAE Reg. No. 2 on assurance of nuclear safety in the course of designing of nuclear installations (1978).
- ČSKAE Reg. No. 4 on assurance of nuclear safety in the course of siting of nuclear installations (1979).
- ČSKAE Reg. No. 6 on assurance of nuclear safety in the course of commissioning and operation of nuclear power installations (1980).

The fundamental basis criteria used at that time included practices of countries with developed nuclear power industry and recommendations of the International Atomic Energy Agency in Vienna.

1.1.1 Legislative environment

The nuclear legislation of the Czech Republic was fundamentally amended in 1997 by issuance of the Atomic act (Act No. 18/1997 Coll., on peaceful utilization of nuclear energy and ionizing radiation) and its implementing decrees.

The basic input document for an application to site a nuclear installation is the Pre-siting Safety Report, which shall include:

1. Description and evidence of suitability of the selected site from the aspect of siting criteria for nuclear installations or radioactive waste repositories as established in a legal implementing regulation;
2. Description and preliminary assessment of a design conception from the aspect of requirements laid down in an implementing regulation for nuclear safety, radiation protection and emergency preparedness;
3. Preliminary assessment of impact of operation of the proposed installation on the personnel, the public and the environment;
4. Proposal of a conception for safe termination of operation;
5. Assessment of quality assurance in the process of selection of site, method of quality assurance for preparatory stage of construction and quality assurance principles for following stages.

A Preliminary Safety Report shall be submitted for the issuance of the construction permit, which shall include:

1. Evidence that the proposed design meets all requirements for nuclear safety, radiation protection and emergency preparedness as laid down in implementing regulations;

2. Safety analyses and analyses of the potential unauthorized handling of nuclear materials and ionizing radiation sources, and an assessment of their consequences for personnel, public and environment;
3. Information on predicted lifetime of nuclear installation or very significant ionizing radiation source;
4. Assessment of nuclear waste generation and management of it during commissioning and operation of the installation or workplace being licensed;
5. Conception of safe termination of operation and decommissioning of the installation or workplace being licensed, including disposal of nuclear waste;
6. Conception for spent nuclear fuel management;
7. Assessment of quality assurance during preparation for construction, method of quality assurance for the carrying out of construction work and principles of quality assurance for linking stages;
8. List of classified equipment.

The requirements for evidence of suitability of the selected site are specified particularly in the SÚJB Decree No. 215/1997 Coll., on criteria for siting of nuclear facilities and very significant ionizing radiation sources. The decree establishes criteria decisive for the assessment of suitability of the selected site of nuclear installations and workplaces with very significant ionizing radiation sources (hereinafter the “siting”) from the viewpoint of nuclear safety and radiation protection and defines eliminating and conditional criteria for assessment of the considered siting of a nuclear installation.

The eliminating criteria are:

- expected exceeding of the specified average annual effective exposure doses of individuals from the critical group of population on the location,
- impossible timely introduction or complete implementation of all urgent measures for the protection of population,
- occurrence of karstic phenomena in the scope threatening stability of the rock massif in the subsoil and rock cover of the selected territory,
- demonstrations of post-volcanic activities,
- achieving or exceeding of the intensity of the maximum computed earthquake of degree 8 of the MSK-64 scale,
- occurrence of geodynamic phenomena, deformation of the surface of the territory as a result of mining activities,
- the site overlaps with areas liable to flooding and many others.

The conditional criteria make it possible to use a territory or a land plot for siting on the condition that a technical solution of unfavorable territorial conditions is possible or available.

The suitability of a selected territory for siting of an installation or workplace , from the viewpoint of the eliminating and conditional criteria in Section 4 and Section 5 of the SÚJB Decree No. 215/1997 Coll., is proved with documented results and analyses of targeted investigations and surveys performed on the given territory, or based on supporting documents, data and information from earlier investigations and surveys kept on file, unless such documents have lost their validity since the time of their development.

Specific requirements for designs of nuclear installations are contained in the SÚJB Decree No. 195/1999 Coll., on basic design criteria for nuclear installations with respect to nuclear safety, radiation protection and emergency preparedness. In respect to external risks the

Decree defines the following requirements for protection against phenomena brought about by natural conditions or human activity outside the nuclear installations:

The Section 10 Protection against phenomena caused by natural conditions or human activities outside the nuclear installation:

i) The components important for nuclear safety of a nuclear installation shall be designed in a such way, that under conditions of such natural events, which could be reasonably considered (earthquakes, windstorms, floods, etc.) or under the events resulting from the human activities outside of nuclear installation (aircraft crash, explosions in the plant vicinity etc.) it should be possible to safely shut down the reactor and to keep it in sub-critical state, to remove the residual power of reactor for a sufficiently long period, and to assure that appropriate radioactive leakage does not exceed the limit values stipulated by the special regulation.
ii) In the design of nuclear installation the following shall be considered:
   - the most important natural phenomena that have been historically reported for the site and its surrounding vicinity, extrapolated with a sufficient margin for the limited accuracy (uncertainties) in values and in time,
   - the combination of the natural phenomena or effects resulting from the human activities and the accident conditions caused by these effects.

In compliance with the IAEA recommendations, the above-mentioned implementing regulation of the Atomic act requires that the design should take into account historically the most serious phenomena recorded in the given location and its surroundings and to extrapolate them to a period up to 10 000 years, including a combination of effects of natural phenomena, phenomena caused by human activities and emergency conditions induced by such phenomena. Based on a probabilistic evaluation some events may be excluded if the probability of their occurrence is very low. The determination of such a limit value for the individual cases is in the competence of SÚJB.

The SÚJB Decree No. 195/1999 Coll. contains a number of other specific technical requirements for systems of reactor cooling, containment, power supplying systems and their backups, including requirements for their functioning under normal and abnormal operating conditions and under accident conditions, while the latter include external events the occurrence of which may be actually expected based on the known history of the location.

1.1.2 Evaluation of safety of NPP sites

In addition to the licensing procedure in connection with the construction of new nuclear installations, the Czech Republic has a long-established practice of periodic safety reviews (PSR) of the nuclear power plants in 10-year intervals. For this purpose SÚJB has issued a guideline “Periodic safety review” BN – JB – 1.2. The periodic safety review evaluates to what extent the systems, structures and components of a nuclear installation, individually and as a whole, including their personnel, correspond to the current safety requirements contained in legal regulations of the Czech Republic, recommendations made by WENRA and IAEA and international practices and to what extent the original design basis, which had been used as a basis for resolutions made by SÚJB about siting, construction and commercial operation of the nuclear installation, remain valid. The output of PSR is a set of measures that seek to maintain or to improve safety in order to ensure a required level of safety of the nuclear installation throughout the entire period of its operation until the following periodic review or until the end of its service life.

The safety review performed both by the operator and by the state regulatory authority (within the periodic safety review and newly also the so-called stress tests for external effects and events) shall include:

- evaluation of design requirements and compliance therewith,
- evaluation of resistance against beyond-design basis conditions (safety margins, diversity, redundancy, physical separation, etc.) and effectiveness of the in-depth protection, including identification of cliff edge effects and potential measures to avoid such effects,
- identification of all means to maintain 3 basic safety functions (reactivity, fuel cooling, confinement of releases) and supporting functions (electric power supply, I&C) and consideration of possibilities to effectively improve the in-depth protection.

The most recent periodic nuclear safety reviews were performed for Dukovany NPP after 20 years of operation in 2006 and 2007 and for Temelín NPP after 10 years of operation in 2008 and 2009. They represented in-depth inspections of the fulfillment of requirements of national and international legislative documents, WENRA reference levels defined in the document “Reactor Safety Reference Levels” and other IAEA international documents (Safety Guides). The comprehensive evaluation within the PSR identified suitable opportunities for the improvement of safety that were also confirmed by results of the stress tests. Most of the measures are now in the stage of implementation or preparation for implementation and they would have been implemented regardless of the subsequent evaluation during the stress tests. The completed PSR anticipates implementation of the approved measures for Dukovany NPP by 2015 and in some justified cases by the next PSR (2018) and for the Temelín NPP by 2018.

SÚJB evaluates Final Reports from PSRs of the individual units, issues positions on the PSR findings and on the list and completeness of corrective measures and, periodically, at the end of each year of operation, reviews the fulfillment of the schedule and content of the corrective measures. It also discusses with the Licensee any potential changes in the schedule for the performance of corrective measures and approves adopted technical and administrative measures.

The fulfillment of some legislative requirements and the evaluation of real risks are supported by probabilistic safety assessments (PSA) for all the units. The first PSA for EDU units 1 through 4 were performed in 1989÷1993. The PSA study of EDU covered the PSA Level 1 for the conditions on power and for a limited set of internal initiation events. The analyses were subsequently expanded with analyses of other types of risks under non-power conditions, including shutdowns, risks of internal fires and floods, falls of heavy loads and risks of external events. The PSA Level 2 was completed in 1998 and subsequently updated, initially in 2002 and then in 2006.

The original probabilistic models have been continually updated to reflect actual designs of the units after all the gradually implemented safety improvements. The updating of the models also included analyses of fire risks, risks of floods and updating of models used by PSA Level 2. The PSA Level 2 currently includes the operation on power, while assessments for non-power operation and shutdown conditions are currently being performed.

The first probabilistic safety assessment for Temelín NPP, units 1 and 2, was completed in 1993÷1996. The PSA study for Temelín NPP included PSA Level 1 for the full power operation and low-power and shutdown conditions, as well as assessment of risks of fires, floods, seismic events and other external events. The study also included PSA Level 2. The PSA for Temelín NPP was updated in 2003, based on the conditions of the power plant at the time of its commissioning. The analyses performed in 2001÷2003 represent the status of knowledge of how the power plant responds to accident conditions based on the current status of the design and the status of operation after a number of completed safety improvements. This enables to evaluate effects of the adopted measures on the safety design of Temelín NPP in terms of the core damage frequency and large early release frequency, and thus to obtain a more realistic estimate of the current level of safety at the time of the commissioning and further operation.

All the mentioned analyses deal with the resistance of the units against external effects and enable to look for potential other solutions to reduce the risks and their unacceptable consequences.

Based on the continually submitted and reviewed safety documents, both sites can be classified as highly stable in respect to seismicity and with minimum influences of extreme climatic effects. The ultimate heat sink in both NPP sites is atmosphere and the cooling is ensured by evaporation from the cooling towers. The water is supplied from water reservoirs on water streams in the proximity of both power plants, which are situated significantly below the NPP levels and therefore their potential damage (e.g. in case of extreme floods, earthquake etc.) will not endanger either of the plants.

1.2 DESCRIPTION OF ACTIVITIES PERFORMED BY THE LICENSEE

1.2.1 Overview of implemented and planned activities

1.2.1.1 Evaluation of seismic risks

The first evaluation of the seismic risks of the NPP sites was performed in 1979. Based on a probabilistic evaluation of a catalogue of historically reported earthquakes it was concluded that the 5.5° MSK-64 magnitude will not be exceeded during the NPP design service life with the probability greater than 90%. The values of seismic risks of both NPP locations were subsequently reevaluated in 1995 in connection with the recommendations made by IAEA mission (Safety Issues).

In the region of Central Europe and on the territory of the Czech Republic there are no tectonic structures that would enable generation of extremely strong earthquakes in the sites of both nuclear power plants, which would be comparable with the disastrous earthquake in Japan on 11 March 2011. In compliance with the IAEA guideline, the level of seismic risks in the locations is defined by the real value of the maximum design earthquake (MDE) with the occurrence time 1 x 10 000 years (SL2). The realistic values of seismic risks correspond approximately to PGA_{hor} = 0.06g (with 95% probability that it will not be exceeded in the time interval of 10 000 years), or 0.05g (with 90% probability that it will not be exceeded in the time interval of 105 years) for the monitored period of 1000 years (SL1).

Three different approaches to assessment of seismic risks of the locations have been used to confirm sufficiency of the maximum design earthquake (SL2):

- Seismostatistic (probabilistic) – it was processed in two variants using the same catalogue of earthquakes but different compositions of focal areas.
- Seismogeologic (seismotectonic) - it used the assumption that seismic focal points were associated with active faults.
- Experimental - also called a "zoneless method", which does not require definition of source zones and their delimitation or determination of seismicity parameters and

their seismic potential. The method is based on measurements of attenuation characteristics on the route epicenter - evaluated structure.

The resulting values have been determined based on a comparison of results of all methodical approaches, using the most conservative values. The application of a combination of the methodic approaches eliminated inaccuracies in the catalogues of earthquakes, generalizations in the diagrams of focal areas and increased reliability of results of the solutions.

The design values of resistance of the equipment and buildings for both NPPs have been determined as follows:

Tab. 1: PGA – Peak ground Acceleration in the horizontal and vertical direction on the level of free ground

DBE	Level	Peak Ground Acceleration (PGA)	Duration	Comparable $I_{stav.}$
MDE	$SL2_{hor}$	0.1 g	4 - 8 s	7°MSK-64
	$SL2_{ver}$	0.07 g	4 - 8 s	
DE	$SL1_{hor}$	0.05 g	4 - 8 s	6°MSK-64
	$SL1_{ver}$	0.035 g	4 - 8 s	

MDE Maximum Design Earthquake or SL2 Earthquake under the IAEA Safety Standards Series No. NS-G-3.3 and NS-G-1.6, which corresponds to SSE – Safe Shutdown Earthquake used by the American terminology.

DE Design Earthquake or SL1 Earthquake under the IAEA Safety Standards Series No. NS-G-3.3 and NS-G-1.6, which corresponds to OBE - Operating Basis Earthquake used by the American terminology.

The seismologic analyses and geologic investigations in the NPP sites have shown that the occurrence of seismic events with $PGA_{hor} = 0.1g$ is not practically possible. An approximation of curves of seismic risks on the location to higher intensity values has shown that the frequency of seismic events with the intensity 0.1g can be estimated smaller than 1×10^{-8} events per year.

The data associated with the evaluation of seismic risks on both sites are regularly updated. The results of the evaluation of seismic risks in ETE are additionally supported by measurements of stations of a local seismologic network for ETE detailed seismic regionalization (DSR), installed in agreement with IAEA recommendations, which has been in permanent operation since 1 September 1991.

The main task of the ETE detailed seismic regionalization network is registration of local microquakes with the magnitude in the interval 1÷3. Seismic phenomena are registered in 4 categories: teleseismic phenomena more than 2000 km away, regional phenomena 200÷2000 km away, nearby phenomena 50÷200 km away and local phenomena less than 50 km away. In addition to tectonic earthquakes, the network of stations also registers induced mining vibrations and industrial blasts.

The hitherto results of the monitoring may be summarized into the following items:

- within the range of 40 km around ETE no earthquake has occurred with the magnitude greater than 1,

- Within the range of 40 km around ETE there were only 9 microearthquakes with the magnitudes in the range 1÷2 and none with a greater magnitude,
- The evaluation of industrial blasts in quarries in the site neighborhood has proved that the network is capable to reliably detect and localize quakes with the magnitudes 1÷3 within the range of 50 km around ETE.

The evaluations of the historical data and the long-term monitoring have shown that the ETE site is seismically very quiet. Results from the DSR network also document correctness of the overall seismic evaluation of the ETE site. The continual evaluation of locations of epicenters of local microearthquakes in many cases indicated their causality with the geological composition of the southern part of the Bohemian massif.

Based on the historical data, the biggest potential effects of an earthquake in the Dukovany NPP site may be expected from earthquakes coming from Alpine focal areas. Analyses that took into account both the magnitude of the potential quakes and the least favorable attenuation of intensity on the route from the focal zones to Dukovany indicate that in pure theory the maximum expected macroseismic intensity in the location is 6^{o} MSK. The calculation of seismic risks has led to the limit value of macroseismic intensity 5.8^{o} MSK, which is not likely to be exceeded in the time interval of 10 000 years.

The interest zone of Dukovany NPP has been continually monitored by the local seismic station Kozének, whose seismic records have been continually evaluated by Energoprůzkum Praha, s.r.o., and by the local seismic station KRUC, whose seismic records have been continually evaluated by the Masaryk University in Brno – Institute of Physics of the Earth.

Until now the performed analyses have confirmed the absence of any cases of local tectonic quakes. The records about the Dukovany municipality even fail to contain any information about observed effects of any earthquake. The nearest local quakes were reported in the area of Jindřichův Hradec, where the epicenter intensities did not exceed 5^{o} MSK-64 and their macroseismic fields failed to reach the Dukovany area.

1.2.1.2 Evaluation of NPP resistance against earthquake

Regardless of a real seismic danger in the location, all safety-important equipment and building structures have been made resistant (or are being made resistant) to the minimum peak ground acceleration in the horizontal direction $PGA_{hor} = 0.1g$.

Note: Currently, a process to increase resistance of all equipment and building structures important for safety has been under way on all EDU units, to withstand the peak ground acceleration 0.1g. By now more than 90% (among others all the technology) of safety important equipment have sufficient qualification documents that demonstrate the seismic resistance and the same modifications are being completed on other equipment (electric part and I&C). It is expected that the improvement of seismic resistance of SSC at EDU will be completed by 2015.

Both NPPs have identified building structures and technological equipment which are necessary for basic safety functions (reactivity control, removal of heat from the reactor core, confinement of ionizing radiation and radionuclides) during earthquake, as well as structures and equipment whose damage or failure during earthquake might cause a secondary risk to other structures and equipment in their proximity which are important for nuclear safety.

For more accurate definition of the seismic effects the technological systems and equipment in the 1st category of seismic resistance has been divided into the following sub-categories:

- subcategory 1a - requires keeping of full functional capability up to and including MDE,

- subcategory 1b - requires only keeping of mechanical strength and tightness up to and including MDE,
- subcategory 1c - the seismic resistance is required only from the viewpoint of potential seismic interaction and particularly keeping of the stabile position up to and including MDE. The objective is to prevent influence on the equipment classified in the categories 1a and 1b.

All building structures and technological equipment in the 1st category of seismic resistance, including safety-important buildings, components, service systems, I&C equipment and electric equipment have been subject to seismic analyses using experiments, computations or indirect evaluations either. The results have been qualification certificates of their resistance against MDE or higher and, if applicable, corrective measures.

A number of regular activities are performed to ensure continuous compliance of the current conditions of the equipment with the design requirements. The activities include:

- maintaining of the seismic qualification of the equipment and buildings,
- walkdown activities to identify the required condition of the equipment and prevention of the damage, accidents and fires or injury of persons and assurance of operation of the equipment while meeting high safety standards,
- in-service inspections and tests of the equipment,
- predictive and corrective maintenance of the equipment.

Extraordinary inspections, performed from the viewpoint of seismic resistance after the events at NPP Fukushima Daiichi in May 2011, identified no serious nonconformities between the actual conditions and the design requirements.

The ETE units are equipped with a seismic monitoring system (SMS). SMS is activated whenever the set-up threshold value of acceleration is exceeded (0.005 g in the horizontal and vertical directions for sensors in the open terrain and on the foundation slab and 0.015 g in the horizontal direction and 0.045 g in the vertical direction for sensors in the containment). At the same time, corresponding alarms in the control room are activated as well. The activation of SMS and earthquake alarms are not associated with any initiation signals into the control or protective systems of the unit. An overall evaluation of the unit condition is required after every seismic event. A controlled unit shutdown is required whenever the design earthquake level has been exceeded or if the design earthquake level has not been exceeded but signs of seismic damage have been detected.

A system for indication of water in the rooms has been implemented for a timely indication of indoor floods, as indirect consequence of earthquake, and if a water level is detected in any room the procedures describe applicable activities to be performed by the NPP personnel.

The so-called stress tests have verified that for earthquakes which may be realistically expected in both sites the performance of none of the three basic safety functions will endangered.

Designs of both NPPs anticipate that a potential serious seismic event might damage seismically non-resistant equipment and buildings, which might lead to disconnection from the power grid and utilities. The electric power supply to ensure the above-mentioned safety functions would be in this case ensured from emergency power sources (DG + accumulator batteries) situated in seismically resistant objects. The operating reserve of Diesel fuel in seismically resistant objects is sufficient for several days of DG operation. More supplies of Diesel oil would be delivered by tank trucks.

Seismic events may result in a loss of pumping stations of raw water from the nearby water reservoirs (pumping station Hněvkovice for ETE, pumping stations Jihlava for EDU), which have been made seismically resistant only for the design earthquake (0.06g). The documented quantity of water at the locations in seismically resistant objects is sufficient for several weeks to remove heat from the shutdown reactors and spent fuel pools.

An analysis was performed of cases of damaged cascade of water dams upstream the Vltava River (ETE) and the Jihlava River (EDU) as a result of seismic events. A potential flood wave from the damaged Lipno I reservoir (dam rupture) hitting the water reservoir Hněvkovice would correspond to the flow rate of 10 000-years water, however, it would not endanger the ETE site due to its elevation. Also a hypothetic breakthrough wave from the water dam Dalešice poses no threat to EDU site due to its elevation.

In case of an extensive destruction of the infrastructure and long-term loss of access to the site (collapse of buildings, damaged roads etc.) the required activities will be provided for by the personnel present at the time of the event origination. Replacement of the personnel would be ensured in cooperation with state administration bodies (IRS, army, etc.). If the technical support center is not accessible the TSC personnel would perform its activities from the main control room or the emergency control room.

1.2.1.3 Evaluation of risks from the viewpoint of floods

The Vltava River is the nearest source of raw service water for ETE. The service water is supplied from the water reservoir Hněvkovice. The main building structures on the ETE site with safety-important equipment are at the spot height 507.30 m above the sea level, i.e. 135 m above the water surface of the Hněvkovice reservoir, which is a part of the Vltava cascade.

The Jihlava River is the nearest water stream and source of raw service water for EDU. The Jihlava River with a system of the water reservoirs Dalešice - Mohelno passes north of the power plant from the northwest to the southeast. The servis water is supplied from the Mohelno reservoir which serves as a compensatory reservoir for the water dam Dalešice.

The EDU complex is situated on a high plain at the altitude 383.5÷389.10 m above the sea level, while the main building objects with safety-important equipment are at the spot height 389.10 m n. m. The crest of the dam of the water reservoir Mohelno is ca. 80 m below the EDU building objects.

Therefore neither site is threatened by flooding from natural or special floods, not even in case of a hypothetic breakthrough wave from ruptured dams in the cascade upstream the rivers of Vltava (ETE) and Jihlava (EDU).

The groundwater level on the ETE site is 10÷12 m below the ground, i.e. approximately 500.0 m above the sea level. Since ETE is situated on a high plain and the ground water is supplied only from rainfall, the groundwater flows from the ETE site in all directions. The objects or rooms with equipment important for nuclear safety are not endangered by the groundwater shallow horizon.

The groundwater level on the EDU location is several meters below the building foundations. The groundwater in this area comes from and is made up nearly exclusively by infiltration of atmospheric rainfall. The water is naturally drained to the north and south to the water streams of Jihlava and Rokytná rivers. The local super elevation of the average groundwater level in some objects is addressed by pumping from underground wells into the sewerage. The objects or rooms with equipment important for nuclear safety are not endangered by the shallow horizon of groundwater.

Both NPPs are situated in locations with normal precipitations without extremes. The basic design measures against flooding of safety-important technology with rainwater include the location of the power plant complex that enables gravitation draining of rainwater, sufficiently sized storm water drainage, altitude of entrances, drive-ins and gates in respect to the surrounding ground level and gradients of the adjoining communications. In respect to water removal, the sites are developed as cascades, while safety-important objects are at the highest spot height and the other objects are at the edges of the location, which enables draining of the complex by natural gravitation, even if the storm water drainage fails. Regular maintenance of the gravitation storm water drainage system will ensure that objects with safety-important equipment will not be endangered by flooding, even in cases of extreme rainfall.

The real one-day totals of rainfall on the ETE site correspond to the level 47.2 mm (100-years total of rainfall) and 88.1 mm for 10 000-years rainfall. The real one-day totals of torrential rains on the EDU site correspond to the level 77 mm (100-years total of rainfall) and 115 mm (total rainfall in 24 hours at 10 000-years maximum).

1.2.1.4 Evaluation of NPP resistance against floods

Temelín NPP

The evaluation of floods from water streams in the proximity of ETE indicates that at the surface level in case of 10 000-years water in the Hněvkovice profile will be ca. 5 m above the maximum level, which will result in flooding of most of the ETE pumping station for raw water. This may be followed by destruction of the dam Hněvkovice. Both those events would prevent standard supplying of raw water to ETE and both ETE units would have to be shut down.

Note: During the biggest flood so far on the Vltava River in 2002 the surface level achieved in the Hněvkovice profile was that of the maximum level considered for this hydraulic structure. The water passed over the Hněvkovice dam in a standard manner and no significant damage was found on the pumping station for ETE or on the dam.

This means that external floods may cause a loss of the pumping station Hněvkovice that supplies raw water for compensation of evaporated water in the process of heat removal into the atmosphere. However, the reserve volume on the site is sufficient to cool down the units to the cold shutdown condition. The water on the site is in the water tanks, tower cooling system and, last but not least, the essential service water circuit may be made up from the drinking water supply mains.

Thus during external floods the water supply for heat removal as a supporting safety function is ensured as well. On the site there is a sufficient supply of water to ensure heat removal from the core and spent fuel in SFSP into the ultimate heat sink for at least 3 x 12.5 days.

Dukovany NPP

The evaluation of floods from water streams in the proximity of EDU indicates that high water on the Jihlava River may result in a flooding of the pumping station for raw water on the Jihlava river (which ensures supply of raw service water for the operation of EDU). In case of rupture of the dam on the upper water reservoir it is necessary to consider flooding of the pumping station for raw water and loss of function of the power plant supply with raw water.

However, the pumping station for raw water is not classified as a safety system and the loss of the mentioned function is solved by operating instructions – by the requirement to shut down all the reactor units. On the site there is a sufficient supply of water to cool down units into the cold shutdown conditions. The water on the site is in the water accumulation tanks, tower cooling system and, last but not least, the essential service water circuit may be made up from the drinking water supply mains. The current supplies of raw water at the power plant to achieve the minimum level for operation of ESW pumps are sufficient for at least 400 hours.

Additional measures for protection of building objects at both nuclear power plants (ETE and EDU) against floods from water streams have not been planned, based on the evaluation described above. Neither location is endangered by floods from the water streams.

Objects with safety-important equipment are resistant against flooding caused by total rainfall for 24 hours at the 10 000-years rainfall maximum. The other objects are resistant against flooding caused by one-day total of rainfall at the 100-years maximum. In cases when the sewerage system is completely unavailable due to clogging of inlets the storm water on both sites is removed by surface draining.

Frequency [number of years]	100	10 000
Daily total of rainfall ETE [mm]	47,2	88,1
Daily total of rainfall EDU [mm]	77	115

There is no risk that objects important for safety on either site (ETE, EDU) may be flooded from the gravitation drainage system, also because of the regular maintenance of the drainage system. Even in case of theoretical occurrence of short intense precipitations the system of passive storm water gravitation drainage is able to drain the water thanks to its high volume.

For the individual objects the design establishes requirements for resistance against accumulated water which make sure that inlets and installation openings prevent the accumulated water from penetrating the building objects (watertight lids, sufficient height of the openings above the maximum level, etc.). Considering the measures that ensure that accumulated water cannot get into the objects with safety-important equipment the basic safety functions will be ensured even in case of extreme rainfall on both sites. In order to ensure protection against floods caused by external factors, many regular activities have been performed to maintain compliance of the condition of the equipment with the design. Periodic inspections and maintenance of storm water drainage system and a shaft cleaning schedule ensure that the design parameters are continually maintained. The technical condition of the drainage routes is inspected once a year and necessary repairs are made on as-needed basis. It includes, e.g. the inspection of racks (grid) and intercepting traps that are repaired or replaced on as-needed basis.

The so-called stress tests included also verification that external floods do not immediately endanger performance of any of the basic safety function. Internal floods, unlike floods caused by external factors, will have mostly a local character or they may be quite simply managed (e.g. by switching off the pumps).

Each site has its Local Fire Rescue Service (LFRS) equipped with firefighting and pumping technology and trained personnel to intervene in any part of the site. LFRS units also have independent means for media pumping – mobile technology – adapted for pumping of water during floods.

1.2.1.5 Evaluation of risks caused by extreme climatic conditions

In the Czech Republic the evaluation of the real exposure of the locations to climatic phenomena (in general) uses statistical data about annual extreme values of relevant meteorological quantities, measured during the period of at least 30 years at meteorological stations in the surrounding regions that have the same character of climatic conditions as the NPP sites. The method of statistical processing is based on the International Atomic Energy Agency (IAEA) Safety Guide NS-G-3.4 “Meteorological Events in Site Evaluation for Nuclear Power Plants“, using the Gumbel distribution.

The design exposure to climatic effects has used frequency of the phenomenon once in 100 years. The calculation of an extreme exposure to climatic effects has used frequency of the phenomenon once in 10 000 years. Objects in the 1st seismic category shall withstand the extreme calculated exposure so that they do not threaten the function of systems important for nuclear safety. The other objects shall withstand the design exposure.

In order to evaluate the resistance of building objects and equipment against the effects of other natural phenomena the following extreme climatic effects have been considered in the licensing documentation:

Wind

The determination of the loads to be considered for gusty wind used the measured annual maximum values of instantaneous wind speeds. The wind speeds established for the frequency of 10 000 years correspond to the extreme wind speeds for tornados that may occur on the territory of the Czech Republic (F2). From the viewpoint of loads the situation is therefore considered satisfactory even in case of the potential occurrence of tornados. In respect to flying objects generated by a tornado it can be that the potential effects are covered by the requirement for resistance of safety important objects against impact of external flying objects.

ETE: The input value for wind loads of objects on the ETE site was determined based on measurement at the station Prague-Ruzyně, i.e. 49 m/s for the frequency of 100 years and 68 m/s for the frequency of 10 000 years.

EDU: The updated meteorological data about gusty winds in the EDU region were obtained from five stations in the proximity recorded in the period of 50 years. The design load used the value of gusty wind 46.2 m/s (frequency 100 years) and for the extreme load the value was 60.6 m/s (frequency 10 000 years).

Snow/Ice

The anticipated loading with snow has been expressed as the snow water content, i.e. the corresponding water column in mm.

ETE: The input value for determination of loading with snow and water precipitations was 92 mm for the frequency of 100 years and 157 mm for the frequency of 10 000 years.

EDU: Extreme snow values have been determined based on data from the nearby meteorological station Hrotovice. For the frequency of 100 years it was 92 mm and for the frequency of 10 000 years it was 157 mm.

Analyses of potential ice formation in water management objects with open water surfaces, with regard to their function (removal of heat from appliances) and thanks to the temperatures of circulating water, have shown that not even extremely low temperatures would lead to formation of a large quantity of ice that would endanger their operation.

The design solution of the water reservoirs (dams) for pumping of service water ensures that intake and discharge of wastewater are ensured even in case of ice formation in those tanks. E.g. the intake from the Hněvkovice reservoir has been designed so that the inlets into the pumping station are at least ca. 8.0 m under the water surface, which ensures trouble free water supply even under extremely low temperatures.

High /Low temperatures

ETE: The input values for extreme load with outdoor temperatures in ETE have been derived from measurements of outdoor temperatures of air by the meteorological stations in Temelín, Tábor and České Budějovice. The resulting conservative temperature value was determined based on measurements in the Tábor station. The temperatures used for determination of loads were the instantaneous temperatures 39.0 °C for the maximum annual air temperature and -32.3 °C for the minimum annual air temperature for the frequency of 100 years and 45.6 °C for the maximum annual air temperature and -45,9 °C for the minimum annual air temperature for the frequency of 10 000 years.

EDU: The input values for extreme load with outdoor temperatures in EDU have been selected from measurements of outdoor air temperatures in the meteorological stations Kuchařovice, Moravské Budějovice and Dukovany. Annual maximums and minimums were considered for the evaluation. For the conditions of extreme "long-term" high temperatures it was considered operation at the temperature 46.2 °C for 6 hours a day. Results of evaluation of a "long-term" low temperature -35.8 °C have shown that the systems for heating and protection against freezing have a sufficient capacity and their operation is ensured to provide for heating under extremely cold conditions.

Note for EDU: A computation check of limit values of resistance of safety-important object at EDU for the conditions of extreme wind and snow was completed in 2009÷2010. The computation of limit values for loads of the objects checked the internal forces in the individual main load bearing components of the structure for the least advantageous combination of loads. In those cases, when the calculated values of resistance were lower than design or extreme loads, the analysis assessed effects on the equipment situated in those objects and on safety functions performed by the equipment.

Combination of a strong wind and high temperatures

The probability of this combination is ca. 2% in summer months (3/12 of a year), while the consequences are not worse than for each of the mentioned events separately (however, with a lower frequency of occurrence).

Combination of a strong wind and extreme snowfall

The conclusions have been the same as for the risks of the previous combination. Moreover, a strong wind does not permit formation of a high layer of snow on roof structures of the objects.

1.2.1.6 Evaluation NPP resistance against extreme climatic conditions

With regard to real resistance of the equipment and buildings, the performance of all basic safety functions is ensured in all cases of worsened climatic conditions.

All safety-important systems of both NPPs are situated in enclosed (robust) building objects or in underground objects and no freezing of operating media may occur.

Extreme climatic conditions (gusty wind, ice formation, extreme temperatures, etc.) might destruct the external network (both 400 and 110 kV), with subsequent decrease of power of the reactors to the level of the NPP house consumption. In case of loss of house consumption both NPPs would be supplied by emergency power supply sources situated in seismically resistant objects, which are also resistant against extreme climatic conditions. The operating reserve of Diesel oil in EDU and ETE objects, which are protected against freezing, is sufficient for several days of operation of DG. It is not possible to anticipate Diesel oil supplying via pipes on technological bridges from the ETE Diesel oil management system. However, more Diesel oil could be delivered by tank trucks.

In case of a combination of extremely high outdoor temperatures and operation of emergency sources (DG) the maximum design temperature of ESW (33 °C) might be exceeded, which is the technical specification value established for DG cooling (at full power) during its long term operation. The DG operation is possible even with the ESW temperature higher than 33 °C, subject to the condition that the power will be reduced adequately to observe the limit temperatures of lubricating oil (ca. 60 °C) and inner circuit coolant (83 °C).

In case of extreme freezing temperatures it is possible to evaluate consequences of water freezing in cooling towers in EDU and in ETE also in the cooling basins with a sprinkler system (CBSS). The ETE cooling towers perform no safety functions and CBSS are protected against freezing by operating measures because heated water will be always circulating to at least one CBRR. The basin could get frozen if the CBSS is shut down (which is a permitted condition) during a long-term outage of the system under extreme freezing temperatures. As long as heated water circulates in CBSS even a combination of extremely low outdoor temperature and the minimum value of residual heat will lead to freezing of CBSS, which would prevent transfer of heat into the ultimate heat sink (atmosphere).

Computations for minimum outdoor temperatures in ETE have demonstrated that if the lowest possible quantity of heat is released in CBSS the water temperature in CBSS will drop in the long-term below the design value. Thermal computations have shown that ice may appear on the surface of CBSS at extremely low temperatures; however, it does not prevent the operation of CBSS. The slope of the ESW pool walls is sufficient to allow movement of the ice on the water surface (rising water level).

Extreme wind is expected to have the most serious impact at the EDU site. The ESW safety system does not have a separate system for removal of heat into the atmosphere and it is connected with cooling towers on which the ESW is cooled by spreading and therefore, due to the existing resistance of the shell, an extreme wind event might lead to a reduction of heat removal capacity of the cooling tower through ESW into the ultimate heat sink simultaneously for all 4 EDU units. A solution of such a limit situation with a loss of the ultimate heat sink would be analogical as in case of station blackout (SBO) described in EOP procedures.

The evaluation also included a fall of the roof structure of the EDU turbine hall as a result of a snow load because of some safety-important equipment (cool-down systems, SG emergency feed water supply, ESW piping, live steam piping, etc.). An extreme snow load is not an immediate phenomenon. Therefore simple organizational or technical measures may be used (continual removal of the show, shelters, covers for important equipment) to eliminate impacts and to ensure performance of the safety functions. For the performance of such preventive activities it is necessary to amend the documentation accordingly.

1.2.1.7 Evaluation NPP resistance against industrial risks and aircraft crash – design basis and methodology

Protection against effects caused by aircraft crash

Dukovany NPP

The space over the nuclear power plant has been declared as a prohibited air flight zone for all flights in the "Aeronautical information publication", the data of which are binding for all users of the air flight space of the Czech Republic.

The power plant is situated in the proximity of the military airport in Náměšť nad Oslavou (ca. 10 km). The space above the nuclear power plant with the radius of 2 km and height 1500 meters is a prohibited zone for air flights.

Probabilistic and deterministic analyses of possibilities and consequences of aircraft crash of various categories have been performed. The analyses have demonstrated that the power plant is sufficiently protected against effects initiated by an impact of the so-called design aircraft, which in terms of the model corresponds to a passenger or military aircraft. The evaluation of protection against effects caused by an aircraft crash was conducted in compliance with IAEA guidelines. Results of the computations have shown that the aircraft crash will not cause impermissible damage of the primary circuit system because the structure of building parts important for nuclear safety is sufficiently resistant against potential effects caused by the aircraft impact. The analyses have also shown that redundant systems for reactor core cooling, in connection with their varied spatial separation and building protection ensure that after a potential aircraft impact the reactor trip systems and systems for the reactor heat removal system will continue to operate.

Temelín NPP

The space above the nuclear power plant with the radius of 2 km and height 1500 meters is a prohibited air flight zone. The ban has been declared in the "Aeronautical information publication". The nearest aviation route is 18 km from the power plant and therefore the aviation traffic has no immediate effect on the nuclear power plant.

The computations have demonstrated that the power plant is protected against effects caused by design aircraft crash. Results of the computations have shown that the aircraft impact will not cause impermissible damage of the primary circuit system because the structure of building parts important for nuclear safety is sufficiently resistant against potential effects caused by the aircraft crash. The analyses have also shown that redundant systems for reactor core cooling, in connection with their varied spatial separation and building protection ensure that after a potential aircraft impact the reactor trip systems and systems for the reactor heat removal system will remain in operation.

Protection against pressure waves from explosions

Analyses have demonstrated that even a potential explosion during transport or storage of hydrogen, that represents a dominant source of potential explosions inside the Dukovany NPP or Temelín NPP sites, will not endanger equipment important for safety which would cause a total failure of their safety functions. All manipulation with hydrogen supply containers, which are situated outside the reactor units, are performed with an increased attention to minimize potential leakage of hydrogen.

Dukovany NPP

A class II road No. 15 passes around EDU in the distance of ca. 500 min the direction Brno, Ivančice, Dukovany, Jaroměřice nad Rokytnou, Moravské Budějovice. Other roads in the close proximity have lower traffic densities. The analyses have shown that even in the little probable case of a potential extraordinary event on a vehicle transporting a dangerous load the safety of the power plant will not be affected in any way.

The power plant complex is connected to a single-track railway from the eastern direction from Moravský Krumlov and Brno. The probability of a railway accident of trains transporting hazardous materials on this track is, now or in the future, practically zero.

In the power plant proximity there are no additional sources of potential external threats.

Temelín NPP

In the proximity of ETE there are three branches of a transit gas pipeline with the diameters 1400 mm, 1000 mm and 800 mm. Their minimum distance is approx. 900 m from the production units of the power plant. The transit lines transport natural gas. The analyses have shown that even the maximum postulated accident of the gas pipeline (simultaneous rupture of all three branches) will not disrupt functions of building objects or functions of the technological equipment. A number of measures has been adopted to reduce probability of the piping accidents and to limit their potential consequences. The measures include additional installation of spherical valves that shorten the section of the piping that may be isolated and also a system for monitoring of natural gas leakage. Computations and analyses performed by professional organizations and research institutions have been positively assessed by SÚJB.

A busy road class II No. 105 has been developed on the southeast edge of the Temelín NPP site, No. 105 from České Budějovice to Týn nad Vltavou. The other roads in the close proximity have lower traffic densities. More than 10 km from the plant there are two sections of roads which are international and used also for transport of hazardous material (ADR). The analyses have shown that even in a very little probable case of an extraordinary event on a vehicle transporting hazardous material the safety of the power plant will not be affected in any way.

The nearest railway track, situated ca. 1.4 km from the power plant, is the local track Číčenice - Týn nad Vltavou with passenger and freight transport. The frequency of passenger traffic is low. The probability of a railway accident of trains transporting hazardous materials on this track is, now or in the future, practically zero.

Protection against intervention by third persons

Designs of both nuclear power plants (EDU and ETE) also anticipate protection against intervention by third persons. Safety systems are backed up and situated in different places and their power supplies are secured in the same manner. In addition to technical provisions, a system of technical, organizational and regime measures has been used to prevent inadmissible interventions by third persons.

1.2.2 Further steps to be taken by the Licensee

1.2.2.1 Measures and opportunities for further improvement of safety

Despite the robustness of designs of both NPPs against effects of extreme climatic conditions, some opportunities have been identified for further improvement of safety, associated with the above-mentioned conclusions of the evaluation. The objective is to additionally increase the level of defence in-depth and thus to strengthen the NPP resistance against earthquake and adverse climatic phenomena. The considered measures may be divided based on their importance into short-term (1 to 3 years) and medium- term (up to 10 years). The proposed measures are summarized in the section 1.4 hereof.

The provision of alternative supplying of Diesel oil for long-term operation DG from tanks at both NPPs ranks among the solutions to be implemented in a short-time horizon. The integration of diverse systems that provide for the ultimate heat sink (to cooling towers), with regard to its complexity, is ranked among the medium term measures. Currently, however, pre-design preparations have been under way to implement those measures. The other measures concerning e.g. finalization of some procedures for management of NPP under extreme conditions have again the nature of measures to be implemented in the short-term.

From the viewpoint of protection against effects of an aircraft crash, the protection against pressure waves from explosions and protection against interventions by third persons, no significant changes of the protection systems or significant technical measures are anticipated in the foreseeable future. Further specification of the risks has been performed gradually and based on the new findings it will be possible to implement additional corrective measures.

1.2.3 Conclusions made by the Licensee

1.2.3.1 Conclusions from the evaluation of Czech NPPs resistance against earthquake

There are no tectonic structures on the territory of the Czech Republic that would enable generation of strong earthquakes in EDU and ETE sites. The evaluations of historical data and long-term monitoring have shown that both sites have been selected well as seismically very quiet.

The complete seismic evaluations of the sites indicate that in the ETE site an earthquake greater than 6.5° MSK-64 (PGA_{hor} = 0.08g) will not occur with 95 % probability and in the EDU site an earthquake greater than 6°MSK-64 (PGA_{hor} = 0.06g) will not occur with 95 % probability. As all the safety-important equipment and building objects have been made resistant (or are being made resistant) to withstand at least 7°MSK-64 (PGA_{hor} = 0.1g), there is a safety margin for the remaining 5 % uncertainty.

The analyses of seismic resistance of objects and selected equipment indicate that the resistance of all safety important equipment and building objects, in which they are situated, significantly exceed the value PGA_{hor} = 0.1g specified for MDE. The differences in the resistance of individual SSC are individual but they contribute to further improvement of the safety margin to ensure safety functions.

1.2.3.2 Conclusions from the evaluation of Czech NPPs resistance against floods

The locations of the sites exclude the risks of natural or special flooding. Internal floods do not represent risks for nuclear safety because they are of local nature and they can be easily managed.

ETE building objects have been designed as resistant against floods for the maximum one-day precipitations total which at the spot height 507.10 m above the sea level create the maximum water level 47.2 mm for 100-years precipitations and 88.1 mm for 10 000-years precipitations.

EDU building objects have been designed as resistant against floods for the maximum one-day precipitations total which at the spot height 389.1 m above the sea level maintain the maximum water level 115 mm which is the total of precipitations for 24 hours at the 10 000-years maximum.

Even the lowest located building objects with safety equipment are situated above the surrounding grade level, while the height margins are around 20%.

The annual distribution of precipitations in the long-term average is characterized by the highest totals in summer months, with the maximum in June (70 mm) and the lowest totals in winter months and the minimum in January (21 mm). Sewerage networks have been designed as sufficient for removal of storm water by gravitation from the NPP grounds into a storm water drainage collector.

LFRS at both locations have mobile technology available which has been adapted to pump water from local flooding above the value of 10 000-years maximums.

1.2.3.3 Conclusions from the evaluation of resistance of Czech NPPs against extreme climatic conditions

The analyses have demonstrated a sufficient resistance against effects of climatic extremes for all buildings, systems and components that ensure fulfillment of basic safety functions.

In case of extremely low temperatures (in case the Diesel oil in ETE piping bridges gets frozen) it might be necessary to transport Diesel oils for long-term operation of DG (longer than 2÷3 days) in tank trucks.

Only extreme wind could have a significant effect on the EDU site. The extreme wind might cause (in an extreme case) a loss of external power supply and potential damage of CT, which may even lead to SBO.

1.2.3.4 Conclusions from the evaluation of resistance of Czech NPPs against other external effects

The measures implemented by now for protection against effects of aircraft crash, pressure waves from explosions and against interventions by third persons have been found sufficient under the current circumstances.

1.3 DESCRIPTION OF ACTIVITIES PERFORMED BY THE STATE REGULATORY AUTHORITY

1.3.1 Overview of implemented and planned activities

The state regulatory authority has issued the SÚJB guideline - on the requirements for the design of nuclear installations BN-JB-1.0. In respect to external effects the guideline includes, inter allia, the following provisions:

- **Design basis**

- The design basis shall be defined and documented, containing in particular:
    - status (condition) of the nuclear installations, their categories and relevant acceptance criteria,
    - specification of functions (particularly safety functions) and requirements for properties of the equipment important for nuclear safety, designed and necessary for

safe management of individual categories of conditions of the nuclear installation and for the fulfillment of safety objectives under the requirements established by the legislation and regulatory authorities,

- specific assumptions and values (acceptance criteria) representing design limits (under which the mentioned functions are fulfilled), as established by the legislation, regulatory authorities, generally accepted practices and/ or derived from designer´s computations, experiments and experience,
- in justified cases also methods of analyses that demonstrate nuclear safety and radiation protection and other supporting information.

- A categorization of postulated initiation events shall be made with regard to the anticipated frequency of their occurrence and seriousness of radiological consequences. Radiation and technical design acceptance criteria shall be specified for each such category so that initiation events with a high frequency of occurrence have only insignificant radiological consequences and events with significant radiological consequences have very low frequency of occurrence.

- Deterministic and probabilistic methods or a combination thereof shall be used to make a list of postulated initiation events which may have a significant impact on safety of nuclear installations, including those that can be caused by internal or external effects generated by natural phenomena and human activities or a combination thereof.

- Nuclear installations shall be designed so that natural phenomena, which cannot be practically excluded (earthquake, windstorms, inundations and floods, extreme external temperatures, extreme temperatures of cooling water, meteorological precipitations in any form, humidity, ice formation, flora and fauna effects, etc.), or events caused by human activities from outside of the nuclear installation, which cannot be practically excluded (aircraft crash, explosions, fires, traffic and industrial accidents in the proximity of the nuclear installation, electromagnetic interference or other effects of technical equipment existing outside the nuclear installation, etc.), do not endanger particularly the basic safety functions.

- The designing process of a nuclear installation shall therefore take into account:
  - properties of the site where the nuclear installation will be located, in conformity with requirements of a special legal regulation,
  - the most serious natural phenomena or events caused by human activities, as historically recorded in the given site and its proximity, extrapolated with regard to the limited accuracy of values and time,
  - a combination of effects of natural phenomena or events caused by human activities and conditions of abnormal operation or emergency conditions caused by such phenomena and events.

- The design basis and the current physical condition of the equipment and its documentation shall be in conformity at all times. The design basis shall be regularly evaluated (e.g. at the periodic safety review) and they shall be revised or amended, or, if applicable, the equipment shall be modified accordingly, if reasonably practicable and justifiable by a significant improvement of safety.

- **Safety assessment**

(27) A comprehensive deterministic and probabilistic process of safety assessment shall be performed, as an iterative, verification and validation activity to review fulfillment of the general safety objective and basic safety principles during the designing process and during all other stages of the lifecycle of the nuclear installation, which shall demonstrate that the design and the designed equipment are capable, within the scope of the design basis, of meeting requirements for nuclear safety and radiation protection during normal and abnormal operation and during accident conditions anticipated in the design.

(28) This assessment process shall be documented to ensure its independent verification and updating with regard to operating experience, new information, existing level of science and technology and method of evaluation. The basic forms of documentation of the assessment process are safety reports whose content and level of information are determined by the State Office for Nuclear Safety.

In the SÚJB regulatory practice the above-mentioned requirements are transformed into binding conditions in SÚJB resolutions relating to the license for the operation. For example, the licenses for operation of EDU units issued in 2005 and 2007 contained the following condition:

“The applicant shall further develop the accident management program, including management of the so-called beyond design accidents, and inform SÚJB about the results every year by the end of the 1st quarter of the following year.“

Similarly, the license for operation of ETE units 1 and 2 from 2004 and 2005 respectively, contain the following condition: “The applicant shall update the procedures for severe accident management guidance (SAMG), including instructions for activities in the control room and technical support center (TSC). SÚJB shall be informed about the updating once a year, by the end of the first quarter of the following year at the latest.“

These requirements have been continually met by both power plants.

1.3.2 Further steps to be taken by the state regulatory authority

The analysis of PSR findings from the assessment of Czech NPPs after the Fukushima event has shown that that there are no acute safety findings and that it is only necessary to consistently continue in the fulfillment of the prepared schedules and to complete or, if applicable, to finalize and to deepen the prepared measures.

SÚJB will make sure that the Licensee implements the applicable measures in agreement with the approved schedules.

1.3.3 Conclusions made by the state regulatory authority

As described above, no problems have been identified requiring to be addressed urgently. The proposed measures have been elaborated to a certain level of detail and a plan and a schedule are available to proceed with their implementation. Activities of the Licensee are monitored against the plan of implementation of corrective measures from PSR or from other evaluations performed within the stress tests.

1.4 FINAL SUMMARY OF CHAPTER 1

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 1.2.1) Activity - Taken? - Ongoing? - Planned?	(Item 1.2.2) Schedule or Milestones for Planned Activities	(Item 1.2.3) Results Available - Yes? - No?	(Item 1.3.1) Activity - Taken? - Ongoing? - Planned?	(Item 1.3.2) Schedule or Milestones for Planned Activities	(Item 1.3.3) Conclusion Available - Yes? - No?
Topic 1 External Events						
Complete the design for improvement of seismic resistance of EDU and ETE SSC	Ongoing	Medium-term	Yes	Ongoing Supervisory activity	Continually	Yes
Inspection and assurance of anchoring of non-seismic equipment	Ongoing	Short-term	Yes	Ongoing Supervisory activity	Continually	Yes
Implementation of measures for diverse means for ultimate heat sink (for CT)	Ongoing proposal of technical solution	Medium-term	No	Ongoing Supervisory activity	Continually	Yes
Alternative replenishment of Diesel oil from a tank truck for long-term DG operation	Ongoing	Short-term	No	Ongoing Supervisory activity	Continually	Ne
Development of a an operating procedure for extreme events (wind, temperature, snow)	Ongoing	Short-term	Yes	Ongoing Supervisory activity	Continually	Yes
Guideline EDMG for the use of alternative means	Ongoing	Medium-term	No	Ongoing Supervisory activity	Long-term	Ne
Assurance of sufficient personnel after extreme events	Ongoing	Short-term	x	Ongoing Supervisory activity	Continually	x
Resistance of objects (LFRS, CPS, MPU etc.) to withstand extreme conditions	Ongoing	Medium-term	Yes	Ongoing Supervisory activity	Continually	Yes
Development of methodology for evaluation of external effects, verification of completed analyses, potential technical	Ongoing	Medium-term	No	Ongoing Supervisory activity	Continually	Ne

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 1.2.1) **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 1.2.2)** **Schedule or Milestones for Planned Activities**	**(Item 1.2.3)** **Results Available** **- Yes?** **- No?**	**(Item 1.3.1)** **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 1.3.2)** **Schedule or Milestones for Planned Activities**	**(Item 1.3.3)** **Conclusion Available** **- Yes?** **- No?**
measures						
Access to objects, availability of heavy technology	Ongoing	Short-term	No	Ongoing Supervisory activity	Continually	x
Alternative means for communication after seismic events	Ongoing	Short-term	Yes	Ongoing Supervisory activity	Continually	Yes
Seismic PSA	Ongoing	Short-term	Yes	Ongoing Supervisory activity	Continually	Yes

2. DESIGN BASIS

2.1 INTRODUCTION

Basic inputs for the evaluation included safety reports, Probabilistic Safety Assessments (PSA), documentation about Periodic Safety Reviews (PSR), procedures for abnormal and accident conditions – EOP, SAMG, IAEA documents, WANO documents and others. At both power plants the inspections were complemented with walkdowns and checks of important systems and equipment to verify their actual conditions.

Results of such inspections and analyses of safety documentation were used to identify weaknesses and to propose potential measures to improve robustness of the power plants. The evaluation of stress tests included all operating modes and conditions of the NPP units. Evaluation of design considered loss of external sources of electric power supply, total loss of electric power supply and loss of the ultimate heat sink. Attention was also paid to “severe accidents”. When evaluating the extreme scenarios of events the evaluators proceeded according a deterministic approach considering assumed gradual failure of all preventive measures.

Characteristics of both NPPs and their sites were evaluated based on information from safety studies, analyses, surveys, historical experience and engineering judgment. The evaluated cases covered simultaneous occurrence of unexpected (beyond-design) and improbable situations and failures whose combinations might lead to a hypothetic accident condition of a particular unit.

2.1.1 Legislative environment

The fundamental legal regulation relevant for this field is the Atomic act and its implementing legal regulations – see chapter 4. The requirements for evaluation of the sites from the viewpoint of external events have been explained in chapter 1.

The SÚJB Decree No. 195/1999 Coll. contains a number of specific technical requirements for reactor cooling systems, containment, energy supplying systems and their backup, including requirements for functioning during normal and abnormal operation and under accident conditions, including external events which may be realistically expected to occur based on the history of the given site.

Provisions about assurance of heat removal and backup of electric power supply are particularly important from the viewpoint of technical content. The following provisions shall apply, among others, to the assurance of heat removal:

Section 25 Residual Heat Removal System

(1) The residual heat removal system shall assure that during the reactor shutdown the design limits of the fuel elements and of the primary circuit must not be exceeded.
(2) The residual heat removal system shall provide the sufficient redundancy of important components of the residual heat removal system, the suitable interconnection, the capabilities for disconnection of parts of system, the leakage detection and the capability of their retaining so that the system will operate reliably even at a single failure of any of its component.

Section 26 Emergency Core Cooling System

The emergency core cooling system shall assure

(1) reliable cooling of the core under the accident conditions caused by a loss of coolant so that

a. the temperatures of fuel cladding do not exceed the values stipulated by design limits,
b. the energy contribution of the chemical reactions (cladding, water, hydrogen release) does not exceed the acceptable value,
c. the changes of the fuel pins, fuel assemblies and reactor internals, which could influence the efficiency of the cooling, do not occur,
d. the residual heat may be removed for a sufficiently long period

(2) its sufficient redundancy, suitable interconnection, possibility of disconnection of parts of system, leakage detection and the capability of their retaining so that the system will operate reliably even at a single failure of any of its component.

The requirements for electric power supply are provided in the following provisions:

Section 29 Power Supply Systems

(1) Outlet of the power output of a nuclear installation and supply of house consumption shall assure that

a) their external and internal failures of power supply may affect the reactor operation and the heat removal systems as little as possible,

b) the plant components important for operation may be powered from two different sources (NPP generator and electricity transmission network).

(2) The electric power supply to the control and protective systems of the primary circuit, the residual heat removal systems, the emergency cooling system and containment system shall, in addition, enable the power supply from a back-up source, i.e. be redundant independently of operation of on-site NPP generators or off-site electricity transmission network. The control and protective systems shall be continuously powered.

Section 30 Redundancy of Power Supply Systems

(1) The systems, which are with regard to the nuclear safety redundant, shall be powered in a such way that their functional independence may be ensured by the way, that the electrical power systems and their sources are mutually independent. If the number of sources is lower than the number of independent systems the design shall demonstrate that this does not reduce their reliability.

(2) If a single failure of powered systems cannot affect their function then a single failure of the electrical system or source is also admitted.

(3) If operational capability of some system is necessary for the nuclear safety assurance then the electric power supply system shall provide the necessary power even at a single failure, without limitations.

Section 31 Emergency Power Sources

(1) Systems which shall be powered without any interruption (the 1st category loads) shall be supplied from the sources that provide the power immediately (batteries with invertors).

(2) Sources and supply systems which are only put into operation after a certain time of duration of accident conditions (the 2nd category loads) shall be put into operation within a time shorter than the start-up period of the 2nd category loads.

(3) The possibility to carry out the functional testing of emergency power sources shall be assured.

In case of severe accidents the containment plays an important role of the last barrier against release of radioactive substances. The SÚJB Decree No. 195/1999 Coll. contains, among others, the following provisions about the containment systems (accident conditions shall mean design basis accidents):

Section 33 Design Principles

(1) The containment system consists of the hermetic envelope dimensioned for all design basis accidents, closing elements, pressure and temperature reduction systems and venting and filtration systems.
(2) The containment system shall assure that its required tightness is kept under the origination of accident conditions and for a sufficiently long time after their termination.
(3) The containment system shall render the required function for maximum pressures and appropriate under-pressures and temperatures of design basis accidents. It is necessary to consider the influence of the pressure and temperature reduction systems within a the hermetic envelope, the influence of other potential power sources, penetrations and access openings, uncertainties of the calculation models, experimental results and operating experience.
(4) The containment system shall meet the requirements for protection against external effects according to Section 10.
(5) The components of the containment system shall assure their functional capability and limit the influence on other systems and components important from the viewpoint of nuclear safety.

Section 35 Inspection of Hermetic Envelope Tightness

Before the nuclear installation is commissioned the hermetic envelope shall undergo a pressure test to demonstrate its integrity at a testing pressure higher than the design pressure.

Section 41 Hermetic Envelope Pressure Reduction and Heat Removal System

(1) The hermetic area shall be provided with the pressure reduction and heat removal system which, together with other systems, shall ensure, after the termination of accident conditions associated with releases of mass and energy, a sufficiently fast reduction of pressure and temperature in the hermetic area, and which shall ensure that their admissible values are not exceeded.
(2) The system shall assure reliability, redundancy and functional diversity of its important components and assure its functional capability, even at a single failure.

Section 42 Other Systems of Protective Envelope

(1) The containment system shall be provided with systems that assure monitoring of fission products and substances which might get into it at origination of accident conditions. These systems shall be capable together with other systems
a) to reduce the volume activity and to adjust fission products composition,
b) to monitor the volume concentrations of explosive substances so that the integrity of hermetic envelope may be assured and so that the amounts of released radionuclides may be reduced.
(2) The important components of these systems shall be redundant so that they may operate at a single failure.

2.2 DESCRIPTION OF ACTIVITIES PERFORMED BY THE LICENSEE

2.2.1 Overview of completed and planned activities

Brief characterization of sites and properties of both operated NPPs

Considering similarities between EDU and ETE designs, some sections will consist of shared descriptions of both power plants. In those parts where EDU and ETE are different separate sections will be dedicated to each plant.

NPP units of VVER type have the characteristic capability to ensure basic safety functions by multiple diverse systems in the modes of normal and abnormal operation and under accident conditions.

Active safety systems of both power plants are redundant 3 x 100% and they are mutually independent and physically separated. Passive safety systems (hydro accumulators inside the containment) are redundant 2 x 100%. Seismic resistance of all redundant safety systems is ensured, including electric power supply, control systems and other auxiliary systems. Emergency/backup sources of electric power supply systems and control systems are mutually independent, physically separated and seismically resistant (they are subject to the same qualification as the safety systems). Designs of both power plants have diverse systems to ensure fulfillment of three basic safety functions:

1) safe reactor shutdown and keeping the reactor in safe shutdown conditions (subcriticality);
2) removal of residual heat from the reactor core and from the spent fuel (residual heat removal system);
3) limitation of release of radioactive substances so that they do not exceed specified limits (barriers and isolation of the containment).

Dukovany NPP

At the EDU site there are 4 reactor units arranged as two double-units. Containments of the individual units of each double-unit are during the operation separated and there is no risk that the atmosphere from one unit may penetrate the other unit. During modes 6 (refueling) and 7 (total removal of fuel from RPV) in one unit the containment is open to the reactor hall that is shared with the adjoining unit. During operation on power the containments of the units are hermetically separated both from each other and from the reactor hall. The spent fuel pools of both units are situated in the reactor hall.

Therefore, in case of an accident during refueling, it is necessary to address the issue of potential spreading of radioactive substances into the shared reactor hall and open containment of the affected unit. The reactors are technologically fully independent; however, many systems and the auxiliary and supporting equipment may be used by the other units. E.g. the electric power supply, cooling water circulation, fire water etc. may be connected to all the units. A similar option applies for essential service water (ESW). On each unit there are 3 independent ESW systems that cool important appliances. Pumps to supply the individual units are situated in separated rooms and they are power supplied from the respective units and systems but they may be used by the adjoining units as well, i.e. units 1 and 2 may share the pumps and, similarly, units 3 and 4 may share the pumps. Under accident conditions the double-unit arrangement of auxiliary systems enables substitution or replenishment of media in the tanks of safety systems of the emergency core cooling (ECCS) from the adjoining unit. If only one unit in the double-unit is affected it is possible to use water supply in the passive emergency system XL (bubbler system) of the adjoining unit, which may represent at least 1000 m^3 of H_3BO_3 solution.

Considering the total number of 4 units on the site arranged close to each other as double-units and considering the independence of electric power supply of the individual units from external and internal sources (including emergency ones), the electric power supply sources of one unit may be conveniently used in case of SBO on the other unit. Due to the shared supporting systems (treatment of spent fuel pool water) it is possible to interconnect the units for emergency make-up. The location of storage pools outside the containment enables simplified access for emergency make-up using other emergency means (firefighting equipment, etc.).

Temelín NPP

Both ETE units are mutually technologically independent and separated in terms of building structures. The equipment shared by both units includes raw water supplying from the Vltava River and CBSS (cooling basins with a sprinkler system) for the transfer of heat from the reactor core (RC), spent fuel storage pools (SFSP) and safety systems of the equipment to remove heat into the atmosphere as the ultimate heat sink. In case of a loss of make-up of raw water to NPP each of the three CBSS is capable of removing all the heat from both units for 12.5 days without making up.

As indicated by analyses of usability of mobile firefighting equipment on the site, it is possible to make up the operated CBSS from the remaining CBSS by re-pumping water with mobile means. Apart from CBSS (passive, seismically resistant objects), all the other technological systems for heat transport are mutually independent and separated in terms of building structures for both units.

Considering the independence of electric power supply of both units from external and internal sources (including emergency ones), the electric power supply sources of one unit may be conveniently used in case of SBO on the other unit. The two shared DGs may be used according to the design for electric power supply of both units on as-needed basis in order to provide energy for a sufficient supply of coolant in SG. The shared DGs may be also used for supplying of system distributors.

Additional shared equipment that may be significant for management of severe accidents is the supply of boric acid solution kept for both units in the auxiliary building. There is an additional amount of 1600 m^3 boric acid solution which is available for both NPP units (the volume is comparable with the quantity of boric acid solution that is available in the containment sump). A spent fuel storage pool (SFSP) is situated in the respective containment of each ETE unit. This arrangement is advantageous as it prevents release of fission products in case of damage of the irradiated fuel kept in SFSP. The disadvantage consists in a complicated access to SFSP in case of emergency making-up using other emergency means (firefighting techniques, etc.) The spent fuel storage pools may be also affected in case of accident of the reactor equipment situated inside the containment and vice versa.

Summary for both sites (ETE and EDU)

The stress tests have identified opportunities for further improvement of resistance of both power plants against extreme effects, using both organizational and technical measures. Those potential measures will be subject to further analyses of effectiveness or to feasibility studies.

The EDU and ETE Licensee has performed targeted evaluation of safety (or safety margins) in each of the areas described below, using a number of accident scenarios with various developments. The Licensee also evaluated time reserves up to the beginning of severe accidents for the most conservative cases. The Licensee has also prepared a list of recommendations for potential improvements based on the identified risks or weak points found at both plants (see the subchapter 2.4).

2.2.1.1 Loss of electric power supply

The reviewed cases included a loss of working and backup sources on the unit and a loss of all AC sources (blackout). Also the worst case was considered – a loss of electric power supply on all /both units simultaneously (SBO) and, from the viewpoint of configuration, also the most conservative situation with one of the units in outage. The Licensee has focused also on the potential loss of the ultimate heat sink (UHS), as well as its combination with a power supply failure for all units at EDU and ETE sites.

2.2.1.2 Loss of cooling

The assessment in the frame of stress tests included both a loss of reactor core cooling prior to fuel damage and options of cooling damaged fuel inside and outside the RPV.

2.2.1.3 Containment integrity

The Licensee considered potential threat to the containment integrity caused by hydrogen (as a result of steam-zircon reaction after the core damage), high pressure (gas overpressure) and focused also on the interaction of the melt with the containment bottom concrete and on limitation of the risk of potential melt through. The Licensee also considered effects of electric a power supply failure (blackout) on containment integrity. A review of measures after the loss of containment integrity for both power plants has been performed, as well.

2.2.1.4 Spent fuel pools

The Licensee analyzed potential scenarios of heat removal (as a result of SBO and loss of UHS) from the spent fuel pools, both before and after the damage of the stored fuel.

2.2.2 Further steps to be taken by the Licensee

As described above, the proposals for potential safety improvements resulting from the stress tests will be subject to more analyses to check their effectiveness. Measures of technical nature, which would require modifications of the existing designs of the power plants, will be further subject to feasibility studies, including proposals of specific design changes (before the implementation they have to be approved by SÚJB).

The first findings and broader conclusions have been summarized in the following chapter. Potential specific measures to improve defence in-depth for the investigated scenarios are provided in the final summary of the chapter, including the time framework (subchapter 2.4).

2.2.3 Conclusions by the Licensee

The individual scenarios analyzed below are linked to the basic description of layouts of both power plants, as provided above.

2.2.3.1 Loss of electric power supply

The electric systems at EDU and ETE have been designed to meet requirements of the engineering-nuclear part and to respect properties of the grids outside the power plants, particularly with regard to safety of operation and production of electric energy. In case of a loss of power supply the safety is ensured by a high level of mutual diversity and independence of working and backup power sources for house consumption, as well as by redundancy and diversity of the so-called uninterruptible power supply systems (UPS) that supply safety-important systems and components and have their own emergency sources. On-site power supply at EDU and ETE has been designed for the individual units.

At EDU and ETE the loss of electric power supply may occur on one or more units. In comparison with a refueling outage, the operation of a unit on power is characterized by a higher design resistance against a loss of electric power supply (additional defence in-depth barriers). Nevertheless, the least favorable case in terms of safety is the loss of electric power supply on all/both units simultaneously. From the viewpoint of potential configuration of available equipment the most conservative case is if one unit is in outage.

In agreement with the basic concept of the engineering-nuclear part (3 redundant and independent divisions of safety systems), there are also 3 redundant and independent systems of secured power supply systems (3x 100 %) available. Each of those systems (EPS) is a supporting system for safety systems of the respective division and its preparedness to perform the safety functions is regularly tested.

The emergency AC sources of EPS for safety systems are three independent (system) emergency DGs connected to the respective switchboards 6 kV for safe power supply. The emergency DC sources are accumulator batteries that are permanently connected to the respective switchboards.

At each EDU unit the emergency power supply sources are three Diesel generators and station accumulator batteries SPSS1, 2 and 3. The uninterrupted power is ensured through rectifiers and inverters.

At ETE the emergency power supply sources (DG, accumulator batteries) are also safety systems (designed for one unit) with redundancy 3 x 100% and there are also shared sources (designed for both units) with redundancy 100% + 100%. Their functionality is independent of availability of working or reserve sources.

The secured (uninterruptible) power supply systems (UPS) of safety systems in each of the EDU and ETE units are independent and mutually separated in terms of the layout (construction partitions, fire compartments), electric installations and control system.

The emergency power supply systems at EDU and ETE are also used to supply systems that are related to nuclear safety and systems not important from the viewpoint of nuclear safety (NSRS) but still important for general safety of persons and expensive equipment; they are designed as two subsystems that mutually back up each other on the 100% + 100% basis.

At EDU the system is UPS 4 and designed as two subsystems (4.1, 4.2). The emergency power supply source of each subsystem is a station accumulator battery with the capacity 2000 Ah, 220 V and an aggregate for uninterrupted power supply. UPS 4.1 is connected to UPS 1, UPS 4.2 is connected to UPS 2.

At ETE the system consists of two shared Diesel generators for both units. The emergency DC sources for UPS of safety-important systems are accumulator batteries designed for each unit separately.

More details about EDU

Each of the Diesel generators (DG) has a Diesel fuel supply in the operating tank for at least 6 hours (4.5 m^3 of fuel, the maximum consumption at the maximum load 0.7m^3/h). Moreover, each DG can use an additional mutually interconnected pair of supply tanks, with the minimum reserve 110 m^3 of fuel. The re-pumping of Diesel fuel from the supply tanks into the operating tank is performed automatically, based on a falling level in the operating tank. The pumps that transport the fuel are supplied by electricity from the respective DG. Diesel fuel re-pumped from the supply tanks (so the total supply of Diesel fuel is 114.5 m^3) ensures the operation of one DG for at least 144 hours (in reality it is ca. 160 hrs), i.e. 6 to 7 days without necessary external fuel replenishment. The lubricating oil supply for DG engines is sufficient for the entire period of DG operation.

More fuel for DG could be obtained by re-pumping from tanks of the other Diesel generators (e.g. those that are not in operation). Considering that only one Diesel generator is expected to operate on a long-term basis for one unit and assuming that dispatching pumps are put into operation, the fuel would be available for 18 to 21 days without any external Diesel fuel supply to EDU.

The stable load of UPS 1, 2, 3 is lower than the nominal power output of DG (2.8 MW). The only limiting factor for a long-term loss of external power supply may be the Diesel fuel supply. However, as indicated above, each of the emergency Diesel generators has a supply of Diesel fuel for at least 6 to 7 days of operation without the need of external supply of fuel. The quality of Diesel fuel is regularly checked and the fuel is replaced for preventive reasons.

More details about ETE

The Diesel generators have their own fuel tanks designed for operation of the emergency Diesel generators at the nominal load for at least 48 hours without any replenishment (in reality the time is even longer) and they are also designed seismically resistant. The tank of each of the two shared DGs at 100% load (they supply both EPS on both units) is designed for ca. 12 hours of operation. Considering the actual quantity of Diesel fuel in the tank the operation of the emergency DG at the nominal load is ensured for ca. 56 hours. Considering the backing-up concept of safety systems with redundancy 3 x 100 % it is possible to gradually use the individual safety divisions and thus to extend the time of uninterrupted electric power supply, without replenishment of Diesel fuel, to ca. 7 days. The lubricating oil supply for DG engines is sufficient for the entire period of DG operation.

All the above mentioned times are based on the assumption of nominal load of the Diesel generator with the power output ca. 5 MW. However, the real DG load (considering the activities under EOPs, where only the equipment in operation is that currently necessary for safe operation of the unit), will be ca. 2.5÷3 MW. This simple operating measure will extend the period of power supply without replenishment of Diesel fuel by additional 40%, to ca. 10 days. Apart from the tanks situated at the Diesel generators, there is also a Diesel fuel management system on the site.

The quality of Diesel fuel is checked regularly once a month and maintained in conformity with applicable requirements. The central Diesel fuel management system (4 tanks, 1000 m^3 each) provide the supply for long-term operation of DGs and it can be used also for other potential mobile Diesel aggregates. For practical purposes of consumption the overall supply maintained in the tanks is ca. 1000 m^3 of Diesel fuel.

As pumps in the Diesel fuel management system are power supplied from switchboards of unessential power supply, in case of a long-term loss of external power supply it is necessary to ensure Diesel fuel replenishment by mobile means. When replenishing Diesel fuel with mobile means it is possible to ensure operation of the minimum necessary number of Diesel generators (one emergency DG for each unit and one shared DG for both units) for an additional period of at least 3 days (considering the real supply of Diesel fuel for ca. 10 days).

All auxiliary systems of the DG engine and generator (fuel intake into the engine, lubrication oil, internal cooling circuit, charge air, starting air) are autonomous and they are independent of supply of external energy during the DG operation. Operability of DGs and their auxiliary systems is regularly checked.

Loss of external power supply

At neither of the power plants the loss of external power supply (e.g. on destruction of the grid accompanied by a simultaneous loss of switchyards 400 kV and 110 kV) will cause a transition to emergency power supply if the unit is operating on power.

If an EDU or ETE unit is disconnected from the external grid 400 kV for external reasons then the turbo generator (TG) and the reactor automatically decrease the power to a level which will in the long-term cover the unit house consumption. *Note on EDU: The long-term operation of TG for the purposes of house consumption has been tested several times and after the TG refurbishment within the project of power increase it was tested again. House transformers provide power supply to four 6 kV switchboards for unit house consumption that supply the main drives of the primary and secondary circuits and also switchboards for uninterrupted power supply 6 kV that supply safety systems drives.*

If this does not happen (the units in outage, TG do not working or neither of them has been regulated or is subject to shutdown) then the situation is considered a loss of working power supply of the unit. In that case the home consumption is automatically switched to a reserve power supply source 110 kV (collective *automatic reserve substitution*), Diesel generators (DGs) do not start, accumulator batteries are charged in a standard mode and ensure uninterrupted power supply of DC power. Only in case that the above-mentioned automatic switch to reserve power supply does not occur then a loss of working and reserve unit power supply occurs, i.e. the so-called total loss of on-site power (LOOP).

The loss of working and reserve power supplies on an EDU and ETE unit causes a reduction of voltage in the switchboards of uninterrupted power supply 6 kV. A signal of loss of on-site power (LOOP) is generated, section switches will switch off and all three DGs of the affected unit will start up. The section switches will disconnect EPS switchboards 6 kV from the switchboards of unsecured power supply 6 kV and thus from the normal power supply grid. After DGs start up and connect to SPSS switchboards 6 kV they will gradually automatically start up safety-important drives under the program of emergency load sequencing gradual loading. While EPS switchboards 6 kV are without voltage the uninterrupted power supply of appliances and distribution of UPS in category 1 is ensured by accumulator batteries.

In case of a total loss of on-site power at EDU or ETE none of the basic safety functions of the power plant is endangered. In the mode of a loss of external power supply the units may be maintained in the hot condition in the long-term or cooled down to the cold condition or safely maintained in the outage mode. The power supply of all necessary engineering and I&C systems is ensured by starting of at least one of the three emergency DGs at each unit (and at ETE at least one for the shared DGs, nevertheless, in order to cool the unit down into the cold condition it is sufficient to start at least one of the three emergency DGs on each unit).

If the unit is on power at the time of a total loss of on-site power then the reactor will trip based on a RTS signal and all Main Coolant Pumps (MCP) will stop. Removal of residual heat from core proceeds under the mode of natural circulation, by removal of steam from SG through the turbine bypass steam dump to the atmosphere. Water make-up to SG is provided by means of two auxiliary feed water pumps, which pump water from the feed water tank (FWT), supplied with pumps for demineralized water 1 MPa from 3x 1000 m^3 tanks (EDU), or auxiliary condensate pumps, either from the turbine condenser or demineralized water supply tanks 2x800 m^3 (ETE).

Alternatively, it is possible to make up water into DG with the emergency feed water pumps (EFWP), which should pump water from the tanks 3 x 1000 m^3 directly into selected SG in case that making up of SG fails from EFWP1, 2 (EDU) or with SG emergency feed water pumps, which should pump water from the tanks 3 x 500 m^3 directly into the selected SG (ETE).

If an EDU unit is shutdown at the time of a total loss of on-site power with water-water mode of cooling down, the making up of water into SG is ensured by cooling down pumps in a closed circuit. Alternatively, the water may be made up into SG by SEFWP, which would pump water from the tanks 3 x 1000 m^3 directly into selected SGs. However, for an open reactor with a low level of coolant at the beginning of Mode 6, it is necessary that both the EFWP are operated to prevent loss of natural circulation.

If an ETE unit is shut down at the time of a total loss of on-site power then the heat from the core is removed with a residual heat removal system. Each of the three cooling circuits contains a circulating pump and heat exchanger. The heat exchangers are cooled by essential service water (ESW). Residual heat removal pumps and ESW pumps are power supplied by DG for EPS of safety systems.

Loss of external power supply, reserve AC sources and emergency power supply (station blackout)

The station blackout (SBO) represents a loss of all external and internal (working, reserve and emergency) AC sources. This means a loss of normal power supply from the switchyard 400 kV, loss of reserve power supply from the switchyard 110 kV, non-regulation of the turbo generator to the house consumption, failure of all emergency DGs for uninterrupted power supply of safety systems and failure of an interconnection with the adjoining unit. This is considered an NPP accident condition.

An event associated with a total loss of AC electric power supply of the blackout type (SBO) at ETE or EDU is classified as a beyond-design basis, highly improbable accident. The most serious mode from the viewpoint of NPP is a simultaneous occurrence of SBO at all units on one site.

In order to manage SBO it is important to know the values of key operating parameters. The values of safety-important parameters are communicated by the post-accident monitoring system (PAMS). The only source of electric energy in the SBO mode is the local UPS. The respective I&C systems, reactivity measurement systems (ExCore, InCore), systems communicating parameter values, as well as the PAMS, are supplied from accumulator batteries of UPS 1, 2, 3. During SBO the accumulator batteries are discharged because no source is available to recharge them. If no alternative sources are used during SBO to recharge the accumulator batteries then their capacity is limited in time.

SBO may occur at EDU only in the case that all the following levels of defence in-depth of electric power supply fail:

- external working sources - standard power supply from the switchyard 400 kV Slavětice,

- internal working sources - failure to control any of the turbo generators to the house consumption,
- external standby sources - standby power supply from the switchyard 110 kV Slavětice,
- external standby sources - standby power supply from the switchyard 110 kV Sokolnice,
- external standby sources - standby power supply from the switchyard 110 kV Čebín,
- internal standby sources - power supply from the house consumption of the twin unit,
- all three redundant emergency AC sources for SPSS 6kV on all 4 EDU units (i.e. 12 DGs in total),
- diverse external AC source of the hydroelectric power plant Dalešice via 110 kV lines,
- diverse external AC source of the hydroelectric power plant Dalešice via 400 kV lines,
- diverse external AC source of the hydroelectric power plant Vranov via 110 kV lines.

SBO may occur at ETE only in the case that all the following levels of defence in-depth of electric power supply fail simultaneously:

- external working sources - standard supply from the switchyard 400 kV Kočín,
- internal working sources - failure to regulate a turbo generator to the house consumption,
- external standby sources - backup power supply from the switchyard 110 kV,
- internal standby sources - power supply from the switchyard 110 kV of the twin unit,
- all three redundant emergency AC sources for EPS of safety systems (emergency DGs) on both units (i.e. 6 DGs in total),
- both emergency AC sources for EPS of safety-related systems (2 shared DGs),
- diverse external AC sources (hydroelectric power plant Lipno and small hydroelectric power plant Hněvkovice).

Management of SBO at EDU

In case of SBO the personnel would have the following options to recover AC power supply from the internal and external sources. Works on the recovery of power supply of safety systems from internal and external sources would be performed simultaneously.

Internal sources:

The anticipated options are the use of autonomous sources of AC electric power supply and possibilities of their simple interconnection via a reserve bus bar 6 kV. If SBO occurs only on some EDU units and at least two DGs would operate on another unit then the unit affected by SBO could be supplied from a DG of another unit, i.e. one of the following options could be used:

- voltage recovery from another EDU unit via the standby bus bar 6 kV,
- voltage recovery from DG via the standby bus bar 6 kV.

External diverse sources (the main strategy to address the loss of AC power supply):

If an option remains to use selected power supply routes 400 kV or 110 kV for EDU after events leading to SBO then the on-site power would be primarily supplied from selected units of the nearby hydroelectric power plants Dalešice and Vranov in agreement with abnormal operating procedures. A precondition for this is the possibility to communicate with the concerned external workplaces (pumped storage power plant Dalešice, hydroelectric power plant Vranov, switchyard Slavětice, central control ČEPS, E.ON). Recovery of power supply from the pumped storage power plant Dalešice (4 x 112,5 MW) or hydroelectric power plant Vranov (3x 6,3 MW) has been repeatedly tested (in 2004 and in 2010) and the option was verified.

After SBO analyses the pumped storage power plant Dalešice was selected as the main external AAC and its function has been practically tested. The power plant Dalešice (power output 4 x 112.5 MW) has the ability of a black-start. The test has verified the ability to provide power supply within 30 minutes (on 400 kV lines) or within 60 minutes (on the 110 kV lines).

If an SBO occurs on a unit in hot condition then the EDU shift supervisor shall announce the EXTREME EMERGENCY which, according to the Code of the Transmission Grid of the Czech Republic, defines the necessity to provide energy from the external grid to the affected unit within 1 hour. If an SBO occurs on a unit in semi-hot condition then the DANGER is announced which means the necessity to provide energy from the external grid to the affected unit within 2 hours.

Management of SBO at ETE

In case of SBO the ETE personnel would have the following options to recover AC power supply from the internal and external sources. Works on the recovery of power supply of safety systems from internal and external sources would be performed simultaneously.

Internal sources:

- Power supply from AC emergency sources for EPS of safety-related systems (the so-called shared DG – the same design as emergency DGs).
- Power supply from the twin unit (when TG has regulated its power to the house consumption).

External diverse sources (main strategies to address the loss of AC sources):

- Power supply from the hydroelectric power plant Lipno (ELI 2 x 60 MW) using dedicated lines. In case of the grid destruction the power may be supplied to ETE from ELI which may start up even without an external source (black-start). In case of the grid destruction the plant may start and the control center will set up the route to supply home consumption for ETE. The time necessary to supply voltage from ELI to ETE is about 30 minutes and this option has been confirmed by a test (verification of organizational measures to manage SBO, function of TSPP systems, function of communication means, roles and procedures for key persons in case of SBO).
- Power supply from small hydropower plant Hněvkovice (2 x 2.2 MW to 2 x 4.8 MW, depending on the water gradient) – the voltage to ETE may be brought via the switchyard Kočín 110 kV on the reserve supply line 110 kV.

At ETE there are additional AC sources which are not intended by the design for supply of safety systems in connection with SBO:

- DG for power supply of lubricating oil pumps of the turbine (power output 200 kW),
- DG for the data center (power output 1 MW).

Even though the option to connect these sources into the existing power supply system is not anticipated in the design or tested, their output is sufficient and they could be used for long-term recharging of the accumulator batteries.

Usable capacity of batteries

The discharging time of batteries of safety systems depends on the current load in the time. The capacity of accumulator batteries UPS 1, 2 and 3 at EDU is 3 x 1500 Ah. For each UPS 4.1 and 4.2 the capacity of accumulator batteries is 2000 Ah and their real operating time without a principal load reduction is about 6 hours. The capacity of accumulator batteries UPS 1, 2 and 3 at ETE is 3 x 1600 Ah (2 hours at the real load). For UPS systems relating to safety the capacity of accumulator batteries is 2 x 2000 Ah (5 hours at the real load) and 2 x 2400 Ah (3 hours at the real load). When taking into account the current status of the accumulator batteries and the actual load and also the potential reduction of the connected appliances it is realistic to expect that the time of operation will be several times longer. Analyses have demonstrated that the even a small reduction of the load extends the time of use. The instructions have been already included in EOP.

Conclusions from the evaluation of resistance of Czech NPPs against loss electric power supply

The EDU and ETE sites have no alternative or mobile sources of AC power (except mobile LFRS electricity stations at EDU) which could be used to address a long-term SBO. Nevertheless, there are external sources whose availability and usability in case of SBO has been verified and tested.

Even though there would have to be a multiple failure of defence in-depth in the NPP electric part before SBO, the consequences of SBO would be so serious that additional measures have been proposed to improve the already very robust design from the viewpoint of assurance of electric power supply for house consumption of safety systems, including the possibility to connect alternative sources into the existing power distribution system and their testing. Further, there are potential opportunities for improvement of the power plant´s resistance against a loss of electric power supply in order to strengthen the level of defence in-depth at both power plants during initiation events beyond the existing design, which might lead to the loss of ability to perform safety functions during SBO:

- alternative means to provide AC power for the existing equipment that assures cooling and removal of heat from reactor core and SFSP, including the possibility to connect them to the existing electric power distribution system,
- diverse means for cooling down and removal of heat from RC and SFSP, including the possibility to connect them to the existing technology,
- alternative means to provide DC power and cooling for I&C systems necessary for monitoring of status and control of selected components,
- alternative means for activities and functional communication (internal and external) of personnel.

2.2.3.2 Loss of cooling

The water systems of both NPPs have sufficient supplies of water for removal of residual heat from spent fuel, either in reactor core or SFSP. At EDU the water supply available is at least for ca. 39 days of removal of residual heat (operation of ESW pumps) from shutdown reactors without replenishment of water into the EDU systems. The removal of residual heat at ETE without external replenishment of water can be ensured for a period of at least 30 days subject to the condition that all safety divisions will be used gradually or that water supply from CBSS of non-operable ESW systems will be re-pumped by mobile means.

Dukovany NPP

The water available for replenishment is demineralized water from the tanks 3x 1000 m^3 for each double unit; according to analyses the quantity would last for 72 hours for all the 4 units. When taking into account the coolant supply in FWT, the coolant available to supply the steam generators of all the four NPP units would last ca. 4 days. Apart from the coolant in the demineralized water tanks, SG may be alternatively supplied by mobile means also from cooling tower pools or other sources. In case of a loss of supply of raw water, as long as unessential power supply is available, it is possible to use coolant in the decantation tanks ca. 5 x 2000 m^3 and supplies of raw water in gravitation reservoirs 4x 2000 m^3 for compensation of ESW losses by evaporation.

Directly on the site there are 3 mobile LFRS pumps (the pressure at the pump delivery is 0.8÷1.2 MPa, flow rate 120÷150 t/h), which may be conveniently used as an alternative manner to make up of demineralized water directly into SG. The design amendment included implementation of connecting points for the purpose. The alternative method of making up of water into SG is described in EOPs, it has been practically tested several times and the capacity of the technology to ensure basic safety functions has been verified. The realistic time to actually deliver water into SG with a mobile pump, after the request is made to activate LFRS, is ca. 20 minutes. In case of a loss of water level measurement in SG and other data, tables have been developed to ensure optimized making-up of demineralized water that indicate the flow rate of demineralized water into SG necessary for the respective backpressure in SG, so that the flow rate of the making-up demineralized water corresponds to the removal of steam via the steam dump to atmosphere.

In case of SBO (black-out) at all four EDU units at the same time the capacity of firefighting equipment may represent a certain limitation (emergency plans have not yet been developed for making-up of SGs of two units with one pump at a time). Another alternative option is to use firefighting equipment to make up evaporated coolant to maintain the temperature of fuel in SFSP. This method of alternative making-up of SFSP is described in EOPs but specific procedures for interventions on the site have not been developed. Moreover, SAMG anticipates the use of portable Diesel aggregates to control some fittings (valves) directly from switchboards but specific procedures for interventions on the site have not been developed.

Temelín NPP

Demineralized water can be made up into SGs from the supply tanks 3 x 500 m^3 of the SG emergency supply system for each unit and also from the tanks 2 x 770 m^3 shared by both units. This water supply is sufficient to cool down the units into the cold condition (according to the design one SG emergency supply system is sufficient to cool down the unit into the cold condition) or to maintain units in hot condition for ca. 72 hours. Each CBSS is capable, even in the most adverse case, remove on a long-term basis all heat from the unit with a shutdown reactor without making-up while the ESW temperature does not significantly exceeds the maximum design value. The most adverse case is the situation with LOCA on one unit and the other unit being shut down, i.e. the source of heat into ESW is at its maximum.

As there are three ESW redundant systems on each unit it has been demonstrated that the removal of heat into the ultimate heat sink can be provided for at least 30 days subject to the condition that all safety divisions will be used gradually or that the water supply from CBSS of non-operable ESW systems will be re-pumped by mobile means into CBSS of the operable ESW system. The alternative means for the transport of media is the LFRS mobile techniques.

Time necessary to ensure electric power supply and to recover reactor core cooling before fuel damage

In case of SBO without an intervention by the personnel the removal of residual heat from an reactor core would be in the mode of natural circulation of primary coolant by automatic removal of steam from SG via safety valves (or steam dump to atmosphere) that keep the set-up pressure in the main steam collector. However, the supply of water into SG is completely interrupted and the level in SG gradually drops and thus reduces the effective heat exchanging surface. Consequently, the capacity of heat removal in II.C decreases. If no activity requested by EOPs is performed then the heat removal from the reactor core by natural circulation via SG would stop and subsequently the temperature and pressure in I.C would increase. After the set-up values of pressure in I.C are reached PRZR SV (or PORV) would open and coolant would be removed from I.C via PRZR SV (or PORV). This would temporarily prevent the temperature in I.C from rising, however, it would also mean a non-compensated loss of coolant from I.C into the SG box and, at the same time, the parameters in the containment would increase. The most limiting factor during SBO is the time for which the unit remains without damage of fuel in the reactor core. Another aspect limiting the time for which the unit may remain in the SBO mode is the time before the accumulator batteries are discharged.

Dukovany NPP

Nevertheless, EOPs provide sufficient instructions to the control room staff to ensure that the removal of heat from I.C is in the mode feed&bleed on the II.C side. The time reserve to prevent the loss of heat removal from I.C is ca. 4 hours. Therefore in the case of SBO the personnel, in compliance with EOP, will use steam dump to atmosphere to depressurize SG. Steam dump to atmosphere are power supplied by accumulator batteries and they may be controlled mechanically (on the spot). After SG is depressurized to ca. 0.7 MPa a spontaneous (gravitation) spill of feedwater from FWT (2 x 150 m^3) occurs into SG and thus also a temporary renewal of heat removal. The heat removal in this mode may be for ca. twenty hours after SBO occurrence. If it is impossible to remove heat from the reactor core by natural circulation via SG then the temperature and pressure in I.C will increase and PORV will open to release coolant into SG boxes (or PRZR SV), while PORV will be also power supplied from accumulator batteries. This would also mean a non-compensated loss of coolant from I.C and, at the same time, the parameters in SG box would increase. After using the entire FWT capacity, another option for the removal of heat in the feed&bleed mode on the II.C side (described in EOP) is the making up of demineralized water directly into SG by an alternative manner, using mobile fire water pumps (the pressure at the pump delivery 0.8÷1.2 MPa, flow rate 120÷150 t/h). In order to improve the EDU resistance against SBO connecting points – fixed hook-up points - have been implemented in all units to interconnect firefighting techniques with the plant technology. An alternative method of SG making-up described in EOPs has been several times practically tested and the capacity of this method to ensure basic safety functions has been verified. In case of SBO at all four EDU units at one time the capacity of the necessary firefighting technology may represent a certain limitation (emergency plans have not yet been developed for making-up of SGs of two units with one pump at a time). If demineralized water is made up at the optimum flow rate then the existing supplies of demineralized water are available from the tanks 3 x 1000 m^3 for each double unit and this quantity has been shown sufficient for 72 hours for all the 4 units. Jointly with the coolant supply in FWT, the steam generators of all four NPP units have the coolant supply for ca. 4 days.

Temelín NPP

As a result of SBO the limiting factor is I&C of the safety systems due to the unavailability of the ultimate heat sink for removal of heat losses from the equipment supplied by accumulator batteries. In case the cooling of this equipment is not renewed then the correct function of I&C equipment could be affected, even though the long-term power supply is ensured. Before the renewal of electric power supply the water supply in v SG ensures heat removal from the reactor core via SG into the atmosphere for several hours. A limiting condition during SBO is the time after which the fuel may overheat in the reactor core. In the most adverse case the limit temperature of 650 ^{o}C could be achieved at the output from the reactor core within several hours after SBO occurs. A similar period of time could be available for renewal of electric power supply in case of a loss of heat removal from RC in a shutdown reactor and reduced level of coolant in the reactor, however, with the possibility of a gradual gravitation flooding of RC from the hydro accumulator (HA). After the loss of heat removal from SFSP there is no immediate threat of overheating of the stored fuel for tens of hours after SBO occurs.

SBO scenarios have been analyzed analytically, as well. In the least favorable case, which is associated with the just shutdown reactor and operation with the water level in the axis of cold nozzles (the so-called mid-loop operation), coolant boiling may occur within ca. 10 minutes after the loss of RC cooling. If no additional measures are taken fuel overheating might occur in ca. 30 minutes. Therefore alternative activities have been proposed and already described in EOP (e.g. draining of HA etc.). In order to ensure removal of heat it is necessary to renew power supply of at least one reserve bus bar by that time, which will prevent uncovering and potential damage of the fuel in an early stage of the accident.

The loss of electric power supply would also cause an interruption of cooling of spent fuel and heating of water in SFSP. The trend of increase of temperature in SFSP after the interruption of cooling depends on the initial conditions (the time since the spent fuel removal from the reactor, quantity of fuel in SFSP, etc.) Even at the maximum thermal load in SFSP there is no threat of damage of the stored spent fuel after the loss heat removal from SFSP and the fuel might be damaged only after tens of hours.

Condition after the damage of fuel in the reactor pressure vessel

Measures for management of accidents that involve a loss of RC cooling after a serious damage of fuel have been described in SAMG strategies for both power plants Dukovany and Temelín.

Dukovany NPP

Apart from the continually increasing temperature, one particular symptom of RC damage by melting is the increase of hydrogen concentration in the containment. Considering the rate of hydrogen generation, it is possible that the concentration of hydrogen might not be manageable quickly enough by the existing recombiners. There is still a time reserve (tens of minutes) for a safe ignition of hydrogen in the initial stage. The analyses have shown that from the transition from EOP to SAMG the time before RPV integrity is damaged by the melt is ca. 7 hours on the condition that all methods of coolant supplying into the vessel have failed. The key strategy used in SAMG in this stage of the accident is the reduction of pressure in the primary system to 1 MPa, particularly to prevent ejection of debris from the vessel under a high pressure. PRZR SV and PORV may be used for the purpose. Considering the high quantity of water in the bottom part of the vessel, the bottom will be damaged only a few hours later. The strategy for renewal of heat removal is addressed in SAMG by means of depressurization and particularly by making-up of I.C. The sooner the water is supplied the better chance there is to stop the accident and to keep the melt in the vessel. Therefore SAMG recommends to start water supply as soon and the source is available and in higher quantities than the minimum flow rate shown in the diagram as necessary for RC flooding. The risk of vessel failure would be significantly reduced by implementation of a strategy that anticipates cooling of the vessel from outside by flooding of the reactor cavity. The success of this strategy has been confirmed by analyses. As a part of implementation of a technical solution to adapt ventilation supply pipes into the reactor cavity, inlet openings from the floor of SG box have been prepared to enable flooding of the reactor cavity room. Measurement of the level had been introduced earlier in the reactor cavity room and drainage into special primary circuit floor drainage system had been closed. Both the described actions support the above-mentioned strategy.

Temelín NPP

The consistently symptomatic approach in SAMG is convenient from the viewpoint of the primary objective – protection of the containment from damage. The SAMG strategies use all available means, if any, to resume RC cooling in order to make up I.C. Each individual system for making-up of I.C is able to deliver a sufficient quantity of coolant to remove residual heat from the damaged fuel. Despite that, the RPV flooding from inside does not guarantee RC cooling because the core may be damaged by melting and its cooling may no more be possible. All the strategies are based on the principle of cooling of damaged fuel inside RPV, i.e. making-up of water into I.C. Due to the thermal output of the reactor and due to the design of the reactor concrete cavity, no possibility has been currently identified for VVER 1000 units with V320 reactors to perform effective cooling of RPV from outside. This fact will be a subject matter of more analyses. If the reactor core is not damaged and it is flooded with water then the cooling of the core is sufficient to prevent its damage. In case there is no water in RC the residual heat is absorbed in RC materials. If the making-up of water into I.C does not start then the heating of RC continues. RC cooling in the stage of severe fuel damage is resumed by activities described in SAMG. The following strategies have been defined to resume RC cooling:

- Making-up of water into a hot dry RC will always positively affect the accident progression. An optimum method to resume making-up of I.C has been specified to minimize subsequent release of fission products into the atmosphere. If the flow rate of the making-up water is sufficient to remove energy faster than the generated residual heat, also the RC cooling can be resumed.
- Another measure after a severe damage of fuel is depressurization of I.C. The purpose of the depressurization is to reduce the pressure in I.C below the value under which no direct heating of the containment may occur because the melted core is not ejected from the reactor under high pressure. There are several methods of I.C depressurization (use of the system of emergency I.C venting, pressurizer relief valve, normal injection into the pressurizer, SG depressurization etc.).

Analysis of SBO scenarios with a loss of heat removal from I.C on the SG side has shown the following:

At EDU, even without performing of alternative activities described in EOPs, there is a relatively long time reserve for renewal of heat removal from I.C. The temperature of 550 °C at the output from RC would be reached in ca. 9 hours after the SBO occurs, if the activities required in the preventive stage under EOP are not performed. Similar time reserves have been found for the “transient“ scenario (a total loss of SG feeding). If alternative making-up of SG is performed in compliance with EOPs then this period may be effectively extended to the order of days. LOCA with a loss of all active systems of the emergency making-up of primary coolant might theoretically lead to an earlier damage of RC. An example of such accidents is the combination of SBO+LOCA. However, PSA studies results have shown extremely low frequencies of such events – less than 10^{-8}/year. Analyses of severe accidents thus focus on more probable LOCA scenarios in which the loss of cooling occurs only in the recirculation stage of operation of the ECCS emergency pumps (transition to sucking from the containment sump). Mostly we can anticipate the that RC damage would be postponed by draining of bubbler condenser trays and so the RC damage would occur much later than in the case of SBO with a failure of alternative methods of SG making-up.

If no alternative activities described in EOPs are performed at ETE there is a very short time reserve to renew the heat removal from I.C. The temperature of 650 °C at the output from RC could be reached in the least favorable case in ca. 2.5 to 3.5 hours after SBO occurrence. The temperature greater than 650 °C at the RC output and continually increasing is considered a “cliff edge“ condition for a severe damage of fuel in RC.

Loss of ultimate heat sink (UHS)

The loss of ultimate heat sink (UHS) is defined as a loss of ability to transfer heat, i.e. loss of function of systems that provide media flow for transferring heat between heat sources and the atmosphere. The unused heat from the unit on power operation or residual heat after the reactor shutdown is transferred into the ultimate heat sink by means of several operating methods:

- using the secondary circuit - the system for condensation and circulation of cooling water – during normal and abnormal operation in the modes of operation on power, TG startup and shutdown and in the emergency mode after the reactor shutdown, provided the working or backup power supply sources are ensured. This method does not make it possible to move the reactor into a cold condition.
- Using the residual heat removal system that transfers heat into essential service water (ESW) – during normal and abnormal operation and under accident conditions, it enables to move the reactor into the cold condition (ca. 50 °C in RC and in SFSP).
- Direct removal of steam into the atmosphere from SG, with simultaneous making-up of SG with feedwater – during abnormal or emergency operation (the so-called secondary feed&bleed); this method does not make it possible to move the reactor into a cold condition (cooling down to max. to ca. 110 °C).
- Alternative method of residual heat removal, feed&bleed method on primary circuit (PRZR SV + ECCS) with heat removal to essential service water– only under accident conditions in case of a loss of secondary circuit equipment. From the viewpoint of the loss of ultimate heat sink this method is equivalent to the residual heat removal into the essential service water system.
- Residual heat from the spent fuel storage pool is removed via exchangers into the ESW system and from there into the atmosphere: at ETE via cooling basins with a sprinkler system (CBSS) and at EDU via cooling towers.

Both EDU and ETE use the system of essential service water to transfer heat into the atmosphere and to cool down I.C into the cold condition, to remove heat from spent fuel in SFSP and the heat from appliances of safety systems and systems related to nuclear safety. Under normal condition all the three ESW systems are in operation at the same time (redundancy 3 x 100%). The ESW system is critical from the viewpoint of assurance of safety and transfer of residual heat, both from fuel in RC and fuel in SFSP into the ultimate heat sink.

For the purposes of evaluation of the loss of ultimate heat sink it is possible to consider a loss of systems of heat removal via the secondary circuit and a loss of the ESW system.

With regard to the redundancy of ESW systems 3 x 100% and additional redundancy 2 x 100% of each ESW division (4 pumps, or 12 pumps in total per EDU double-unit and 6 pumps per one ETE unit), the loss of ability to transfer heat from the sources is conditional on unavailability of all ESW pumps. With regard to the spatial separation of the systems and pumps, the independence of electric power supply and other supporting systems, the simultaneous unavailability of all ESW pumps is extremely improbable. Even in case of operation of only one pump in one division of the ESW system it is possible to ensure fulfillment of the basic safety functions. The only possible cause of the loss of all ESW pumps might be SBO. The alternative method for transport of media at both power plants is the employment of fire brigade mobile techniques.

More details about EDU

The EDU design anticipates a loss of supply of raw water into the circulating cooling water volume. In this case it is assumed that the removal of heat from the ESW system may be ensured into the atmosphere by spreading on cooling tower using the water supplies in EDU. Moreover, it is possible to use the supply of coolant in the decantation tanks ca. 5 x 2000 m^3 and supplies of raw water and water in gravitation reservoirs 4 x 2000 m^3 to compensate losses of ESW by evaporation.

The loss of cooling towers function (spreading of ESW on cooling tower will not be available) with the option to make up raw water from pumping station Jihlava (PSJ) is not critical for EDU, as long as ESW retains its ability to supply water to appliances. The making-up of raw water will enable to perform cooling with ESW without a time limitation. However, the system for making-up of raw water is not a safety system which means it may not be necessarily available in case of LOOP. The analyses of a failure of PSJ and ESW indicate that EDU water systems have a water supply, based on a conservative estimate, for ca. 26 days for the production and making-up of demineralized water, and for ca. 39 days of residual heat removal (operation of ESW pumps) from shutdown reactors, without the supply of raw water. The function of heat transfer into the ultimate heat sink is therefore not immediately endangered. Provided that spreading of ESW in CT cannot be used for water cooling in the cooling down modes of the units then the heat will be accumulated in the water volumes available.

Provided water is supplied into ESW intake sumps at EDU then the acceptable temperature of ESW can be maintained for more than 72 hours. However, in case of LOOP it will not be possible to anticipate supply of raw water from PSJ. Without the making-up of cold water the temperature of the service water will increase.

When using water from decantation tanks, additional 3 hours may be achieved for one ESW division. The increased ESW temperatures would lead to limitation of DG cooling but this can be successfully compensated by a proportional reduction of the DG load. If this is not done and if no other measures are adopted (increased ventilation of the room, mobile air-conditioning unit) the overheating of DG might lead to gradual shutdown of DG. A total loss of ESW does not mean an immediate problem (see also SBO) with regard to the option of long-term removal of residual heat into the atmosphere via SG after the unit is shut down.

The main non-technology means that can be used in case of a loss of ultimate heat sink include particularly pumping technology used by the EDU local fire rescue unit. Apart from LFRS on the EDU site, there are no other alternative or mobile sources available to ensure water circulation or removal of heat from ESW appliances, which could be used for improvement of the response to the loss of ultimate heat sink.

In the outage mode (with an open reactor), when the heat removal from RC is dependent on ESW (cooling in the natural circulation mode, removal of heat on the exchanger of the cooling down system in the water-water mode on the secondary circuit), the loss of ESW will cause an increase of temperature in RC. In this case it is possible to start filling refueling pools (SFSP) by emergency making-up systems of the primary circuits with cold water from ECCS (the volume available is up to 1240 m^3 of boric acid solution - depending on the condition of the technology during refueling) and thus to postpone the temperature increase. The removal of heat from RC can be in this mode maintained for more than 72 hours. If the heat removal via ESW is not resumed the temperature in ECCS tanks and SFSP may increase up to the saturation limit. Further, the safe condition of RC is maintained by a very efficient strategy, keeping the level in the open reactor by gravitation filling with a coolant from bubbler condenser trays. The supply to make up for the evaporated coolant is ca. 12 days.

At EDU demineralized water may be supplied into SG by an alternative method by means of mobile pumps of the fire brigade (the pressure at the pump delivery is 0.8÷1.2 MPa, flow rate 120÷150 t/h). The amendment to the design includes hook-up points that make it possible to interconnect firefighting techniques with the plant technology. An alternative method to make up SG is described in EOPs; the method has been practically tested several times and the capacity of this method to ensure basic safety functions has been verified.

More details about ETE

Heat from each ESW system is removed into a separate CBSS and transferred into the atmosphere by evaporation of water from the water surface and by water sprayed out from jets. Heat from spent fuel in SFSP may be alternatively removed by the containment spray system via a dedicated line for making-up of SFSP and further by evaporation into the containment.

The loss of ultimate heat sink creates less risk for heat removal from RC in the regimes with a closed reactor (all unit modes, apart from outage for refueling) thanks to the possibility to remove heat via SG.

In the shutdown mode with an open reactor the heat removal via SG is no more effective and it creates similar risks both for the fuel in the reactor and in the storage pools. The water for ETE process purposes is taken from the Hněvkovice reservoir with 6 vertical pumping sets. The water from the pumping station is delivered into the water tanks 2 x 15.000 m^3 at the plant with two discharge mains and in case of failure of one discharge main fails the other is able to deliver the guaranteed quantity of 3.4 $m^3.s^{-1}$ with 4 pumps running.

The least favorable case at ETE is situation with LOCA on one unit while the other unit is being shut down, i.e. the source of heat into ESW is at its maximum. Since there are three redundant ESW systems it is possible to demonstrate that the removal of heat into the ultimate heat sink can be ensured, without external making-up of water, for at least 30 days on the condition that all safety divisions will be gradually used or that the supply from CBSS of non-operable ESW systems will be re-pumped with mobile means to CBSS of the operable ESW system.

Even in case of a total loss of ESW the removal of heat from RC may be ensured in the hot condition by systems of normal operation which are not dependent on operation of the ESW system - making-up of SG with auxiliary feedwater pumps and removal of steam into the condenser or atmosphere. Main non-technology means that can be used in case of a loss of ultimate heat sink include LFRS pumping techniques. These techniques, however, has not been considered for mitigation of consequences of technology failures. Apart from these techniques, ETE site has no other alternative or mobile sources to ensure circulation or heat removal from ESW appliances that could be used to address the loss of ultimate heat sink. Computations have demonstrated that one CBSS is able to remove all heat from both units for 12.5 days without making-up. In order to meet the requirement to remove heat for at least 30 days it is possible to re-pump water from CBSS of non-operable ESW systems with mobile means into CBSS of an operable ESW system. Analyses of usability of mobile firefighting techniques have shown that they can be used to re-pump water between CBSS.

During a unit outage (with an open reactor) the heat removal from RC depends on ESW operation. A loss of ESW results in an increase of temperature in RC. In this case it is possible to fill in the pools for wet transport. Without heat removal the temperature in the pools for wet transport will increase to the saturation limit.

Heat can be removed in the long-term as long as the evaporation is compensated by making up. From the long-term perspective, the operation of the ESW system has to be recovered at least in one safety division in order to cool the unit to the cold condition. In case of an ESW loss the situation for heat removal from SFSP is the same as in case of SBO, i.e. interrupted cooling of spent fuel and heating of water in SFSP. The trend of increasing temperature in SFSP after the cooling is interrupted depends on the initial conditions (the time since the removal of spent fuel from the reactor, quantity of fuel in SFSP, etc.)

The use of mobile techniques for process purposes at ETE has not been yet described in the procedures - it is necessary to verify its capacity to ensure basic safety functions and preparedness of hook-up points to interconnect the techniques with the process technology.

Loss of ultimate heat sink combined with a total loss of external power supply (SBO)

During SBO events the ESW pumps are not power supplied. Because ESW is the medium that transfers heat from RC, from spent fuel in SFSP and from components of safety systems into the atmosphere, SBO also means a loss of forced heat removal from I.C and SFSP into the atmosphere. An SBO event on a twin-unit (EDU)/unit (ETE) therefore automatically means a loss of ultimate heat sink of the given double-unit due to the loss of electric power supply to ESW pumps. The simultaneous loss of ultimate heat sink at an EDU twin-unit /ETE unit and loss of electric power supply from working and reserve sources means a loss of DG cooling and thus the SBO situation at the given double-unit. The reason is the mutual dependence between DG and ESW – a failure of one system will cause a failure of both of them.

On the other hand, the loss of ultimate heat sink alone will not affect on-site power supply at EDU or ETE, as long as the power supply is ensured from working or reserve sources. However, if the loss of ultimate heat sink occurs simultaneously with a loss of external power supply and TG fails to regulate to the unit house consumption on at least one of units of the EDU twin-unit/ETE unit then emergency power sources (DG) will start up. After DG connection to the switchboard EPS category II and after its loading the coolant of the DG circuit and lubrication oil would start warming up. In case of a gradual loss of ultimate heat sink the temperatures may be kept down by proportionally reduction of the DG load. In case of a sudden loss of ESW the Diesel generators will overheat and will become unavailable. The power supply of safety systems will be then ensured only from accumulator batteries.

More details about EDU

The feed&bleed strategy on the II.C is available in order to remove heat from affected units during SBO. The strategy is based on the option to supply water into SG by gravitation from

FWT and subsequently with LFRS means and to remove heat from I.C by evaporation of coolant from SG and the generated steam via SBSA into the atmosphere. The ability to remove heat from SFSP is lost completely, with the exception of the option to make up evaporated water with LFRS means. As no additional risks have been identified for the combination of a loss of ultimate heat sink and SBO, the conclusions provided in the chapter on SBO shall apply.

At each twin-unit there are three DGs available for odd units and three DGs for even units that need ESW flow rate for its operation. For DG to operate it is necessary to maintain the temperature of lubricating oil (ca. 60 °C) and temperature of the coolant in the internal circuit (83 °C). In exceptional cases, e.g. at the time of emergency supplying of NPP, grid destruction, loss of on-site power, etc., when it was impossible to replace DG with another DG or another source, the Diesel generator should be operated with de-blocked protections, while the only functional protection would be the protection against the loss of oil pressure. Without the oil the DG engine would seize up and it would not be able to provide power supply even after the ESW supply is recovered. After the Diesel generator starts in the 10th second within the program of gradual startup the electric power supply will be recovered for two ESW pumps (of the respective unit and division). If the ESW flow rate is not recovered then DG cannot be operated in the long-term.

The functionality of the ESW system depends on integrity/functionality of CT. The loss of CT function leads to reduction of the ability to remove heat via ESW into ultimate heat sink. The increase of ESW temperature might lead to a gradual loss of all DGs. The problem might only occur in case of a simultaneous occurrence of LOOP which may gradually lead to SBO. The reason is the mutual dependence between DG and ESW – a failure of one will cause the loss of both of them.

If only SBO occurs on one unit from the twin-unit then the loss of ultimate heat sink may not occur as the ESW units of the adjoining unit will remain operable. The simultaneous cooling of both units for which the ESW system is sized is not anticipated during SBO and so the capacity of the remaining two ESW pumps is sufficient to ensure removal of heat from the unit affected by SBO. However, it is difficult to use the option to maintain ESW flow to appliances (ECCS coolers, SFSP coolers, process condensers) due to the failure of pumps (pumps that ensure normal or emergency heat removal which are necessary to maintain forced flow of the medium for heat removal into the ultimate heat sink on I.C or II.C sides).

Selected devices for heat removal from both I.C and SFSP (cooling pumps for SFSP or RHR pumps) may be alternatively supplied from the adjoining unit (the method of power supply recovery is described in valid EOPs) and therefore there is a realistic possibility that the heat removal will continue, both from I.C and SFSP, and that safety functions will be performed in the long-term. All I&C systems are not in operation in emergency conditions after the reactor shutdown and therefore the generation of residual heat from them will be lower. This will significantly reduce the cooling demand of necessary I&C systems. The most important is PAMS with its own cooling supplied also from the 1st category power supply. An SBO event at both units of a double-unit always means a loss of all ESW pumps on that double-unit and therefore also a loss of the medium that removes residual heat from coolers in I.C and II.C of affected units into the atmosphere. However, the remaining strategy available for heat removal from RC is to make up SG by gravitation from FWT and with LFRS means and to remove vapor from SG via SBSA.

For SFSP in this mode there is no long-term method available to remove heat. If heat removal is not recovered the coolant in SFSP might start boiling and fuel could become uncovered in an early stage of the accident (for more description see SAMG and the chapters on management of severe accidents in this report). It is again possible to use the option to maintain the SFSP level by a gravitation flow from bubbler tower trays. The supply of coolant to make up for the evaporated coolant is ca. for 13 days. An alternative option to make up for the evaporated coolant and to maintain the temperature of fuel in SFSP is mobile firefighting techniques. The option of alternative making-up of SFSP is provided is EOPs but specific procedure for interventions on the site have not been developed yet.

More details about ETE

In ETE it is possible to use heat removal from RC via the secondary circuit (SG) until the water supply in SG is used up only in hot and semi-hot condition of the unit. Nevertheless, currently there is no backup method to remove heat from the spent fuel in SFSP. The above-mentioned indicates that operability of the ESW system for heat transfer into UHS and operability of emergency sources of electric power supply are interconnected.

Pumps of the ESW system that ensure transfer of heat from the sources into ultimate heat sink are power supplied from the secured power supply. In case of SBO there is always the loss of ESW. In this case it is possible to remove heat from RC using the water supply from SG directly into the atmosphere, which means that the immediate loss of ultimate heat sink does not occur. The loss of ability to remove heat from spent fuel in SFSP would occur only in the late stage of the accident. However, the loss of ESW at SBO will limit the time for which important parameters of the unit and NPP are available. Thermal loss from I&C equipment supplied from accumulator batteries without cooling functions as a result of unavailability of ESW systems will cause an increase of temperatures in I&C rooms and subsequent loss of respective I&C systems.

It is necessary to verify capacity of mobile firefighting techniques for transport of media and to select hook-up points for interconnection with the technology to ensure basic safety functions. For pumping and transport of water the local fire rescue brigade has 4 fire trucks, 1 combined firefighting truck and 3 fire engines trailers with the overall nominal output 280 l/s. The use of these techniques for process purposes has not yet been described.

Conclusions on adequacy of the protection against a loss of ultimate heat sink

At EDU and ETE the ultimate heat sink is the surrounding atmosphere. The transfer of unused heat during the unit power operation or residual heat after the reactor shutdown is ensured by the ESW system.

At EDU the water supply is for ca. 39 days of operation of the ESW system for removal of residual heat from shutdown EDU reactors without external making-up of water into the ESW system. 12 ESW pumps in total are available for one main production building (2 reactor units).

At ETE the water supply in CBSS is sufficient for ca. 30 days of operation of the ESW system for removal of residual heat from shutdown reactors without external making-up of water into the ESW system. 6 ESW pumps in total are available for one unit.

The loss ability to remove heat is at both power plants (EDU and ETE) connected with unavailability of all ESW pumps. Considering the spatial separation of the systems and pumps and the independence of electric power supply and other supporting systems, the concurrent unavailability of all ESW pumps is extremely unlikely. Even if only one pump in one division of the ESW system remains operable it is possible to ensure fulfillment of the basic safety functions. The only possible cause of a loss of all ESW pumps could be SBO.

In order to remove heat from RC in a unit in hot or semi-hot condition in case of a loss of ESW at both power plants (EDU and ETE) it is possible to use direct removal of heat into the atmosphere via SGs, which are independent of the heat removal via the ESW system.

Consequences of a non-solved long-term loss of ability to remove heat into the ultimate heat sink, both at EDU and ETE, might be the following in the most extreme case:

- Damage of fuel in RC and spent fuel stored in SFSP as a result of absence of alternative methods of heat removal from RC, SFSP and components cooled with ESW (if the boiled out coolant cannot be made up with LFRS technology).
- A loss of cooling for AC emergency power supply (DG) in case of LOOP may cause SBO.
- Release of radioactive substances during boiling in an open reactor during shutdown (including SFSP at EDU) into the surrounding space.
- Loss of ability to control systems and components and loss of communication to communicate values of important parameters as a result of lost function of I&C systems because it is impossible to remove thermal losses from the I&C of the equipment.

Potential measures to improve the plant´s resistance against a loss of ultimate heat sink

Even though a total loss of ability to remove heat into the ultimate heat sink must have been preceded by a multiple failure of the defence in-depth levels, due to the severity of consequences of such a condition some opportunities have been identified to improve further the design, which is already fairly robust from the viewpoint of assurance of heat removal into the atmosphere as the ultimate heat sink. The purpose of the measures is to strengthen the levels of defence in-depth for beyond-design basis initiation events (earthquake, floods, extreme conditions, results of human activities, etc.), which may result in a loss of UHS:

- diverse means for cooling and heat removal from RC and SFSP, including the possibility to connect them to the existing technology,
- alternative use of the diverse means (proposed under the paragraph 1) – the so-called Extensive Damage Mitigation Guideline (EDMG) with the objective to ensure cooling and heat removal from RC and SFSP,
- alternative means to ensure cooling of I&C systems at ETE, as necessary for monitoring and control of selected components.

2.2.3.3 Containment integrity

Loss of UHS

Dukovany NPP

In mode with a closed reactor the containment integrity cannot be threatened by a loss ultimate heat sink alone. The containment starts heating up but it cannot get pressurized to values which would threaten its integrity (design absolute pressure 250 kPa). Cooling of the containment can be ensured with containment ventilation systems with coolers connected to a chilled water system – the system used for distribution of cold water (ca. 10 °C) for the ventilation and heating equipment of the entire plant.

In case of an accident with coolant leakage from I.C into the containment its integrity is ensured initially with spray pumps, as long as they suck from the ECCS tank. After the spray pumps are switched off to suck from the containment floor the effectiveness of spraying starts decreasing due to the growing temperature at the intake.

In case that spray pumps are not functional then a passive containment spraying system using the vacuum-bubbler system is available. The situation is different in case of a loss ultimate heat sink with an open reactor (during an outage for refueling), when no other barrier is available to stop ionizing radiation and radionuclides. In that case, if the temperature is maintained at the saturation limit, there is a risk of release of radioactive substances from the coolant in SFSP into the reactor hall and potentially also outside NPP.

Temelín NPP

Under normal operating conditions and under abnormal conditions the heat from the containment is removed by means of ventilation systems cooled with ESW. When the temperature increases the heat may be removed from the containment with the chilled water system. A long-term loss of ESW and unavailability of the chilled water system will lead to a loss heat removal from the containment. The temperature in the containment will be gradually growing, however, a colder air will continue to be supplied from the external environment and air extraction system will continue to maintain a negative pressure. The integrity of a closed containment, thanks to its design and ability to withstand high temperatures and pressure effects, might be endangered only in a late stage of the accident. The integrity of an open containment (particularly in situations with the open reactor) may lead to a release of radioactive substances from the coolant if the water temperature is maintained at the saturation limit outside CTMT, due to the absence of procedures for a timely closing of the containment.

To ensure safety functions I&C systems have to function and it is essential to know values of unit key parameters. After a loss of ESW both the respective I&C systems and PAMS will be affected by increased temperatures in I&C rooms. The heat removal from rooms with I&C for safety systems may be alternatively ensured with non-essential service water. It is a system of cooling water for non-essential (non-system) appliances supplied from the non-secured power supply system. This option has been described and used as a backup source in case of a planned shutdown of ESW and it increases the resistance of safety functions performance in case of a loss of ESW, as long as the standard electric power supply is available.

Solution of hydrogen risks inside the containment

Dukovany NPP

In the initial stage of a severe accident the containment integrity is most threatened by an extensive fire or detonation due to hydrogen, followed by a failure of the double-door in the reactor cavity. In the late stage of an accident it is threatened by penetration of debris into the cavity. The containment may be threatened by hydrogen at the beginning of RC damage during the steam-zircon reaction. Due to the large cladding surface and exothermic character of the reaction the hydrogen generation is very fast, i.e. from 0.5 to 1 kg/s. With respect to the speed of the hydrogen generation prior to the loss of geometry the quantity of hydrogen cannot be managed with the existing recombiners. The hydrogen generation would continue even in the late stage during the reaction of the melt with the concrete on the cavity bottom; however the rate would be by two orders of magnitude lower (less than 0.01 kg/s).

From the point of view of the containment's integrity's risk due to hydrogen, the risk in the late stage would certainly increase, subject to the condition that the containment is not disintegrated by that time. It is very likely that a large quantity of hydrogen can be burnt already at an early stage and in a worse case fast combustion or detonation might occur, which would lead to an irreversible damage of the containment and hydrogen would be released without any limitations.

The EDU unit containments are equipped with the system of disposal of post-accidental hydrogen, intended for design basis accidents. For the LOCA design basis accident, when only very small amount of hydrogen is generated, disposal thereof is performed by by means of 17 recombiners situated in the containment. After the Periodic Safety Review in 2006 it was decided to make the EDU design more resistant against severe accidents.

In the final stage of preparations is a project to develop a system for effective recombination of post-accident hydrogen, which will be capable of managing even the hypothetically worst case of a severe accident (in terms of hydrogen generation). The analyses completed by now and the experience from other VVER plants have confirmed that the system made up of high-performance recombiners (ca. 30 pcs) complemented with igniters, in case of spray functioning, may reduce the risk of flame acceleration and exclude the risk of a transition to detonation.

The threat to containment integrity by hydrogen burning is addressed by SAMG, using either the principle of intentional ignition or containment inertization. In order to fully consume oxygen in the part of the containment without bubbler condenser air traps it is sufficient to burn or to recombine ca. 700 kg of hydrogen. Additional hydrogen, generated particularly from the interaction with concrete, only increases pressure in the containment but it does not contribute to the risks of hydrogen burning (because no oxygen is available).

The instructions contain a list of equipment which the MCR staff would attempt to manipulate (change positions of the armatures) in order to generate sparks. In order to inertize the containment it is possible to use a limited discharge of nitrogen from the hydro accumulator; for effective intertization in the current condition of the design it is possible to use water steam which will put off the risk of burning to even higher concentrations of hydrogen. First of all however, the hydrogen will be ignited with the existing recombiners provided its concentration exceeds 10 % in the place of their installation. The existing recombiners therefore do not resolve the hydrogen risks during a severe accident because they may remove only several kilograms of hydrogen in the early stage of the accident. If I.C is depressurized before the core damage (which is currently performed according to EOPs) and the procedure is continued after the core damage then the risk of detonation is postponed and it is limited only to the bubbler condenser shaft.

Temelín NPP

The following two modes of hydrogen burning are the most dangerous for containment integrity – fast deflagration and transition from a fast deflagration to detonation. In order to assess the risks, analysis were made of time courses of spreading and distribution of hydrogen generated during severe accidents in the entire containment. The ETE unit containments are equipped with a system for liquidation of post-accident hydrogen intended for design basis accidents.

This system contains passive autocatalytic recombiners and it is able to ensure a long-term liquidation of hydrogen released during accidents and under post-accident conditions and thus to maintain the hydrogen concentrations at values which may not lead to the ignition. The existing system for hydrogen liquidation may not be sufficient for severe accident. At present designing preparations have been under way for installation of a liquidation system for hydrogen generated even during severe accidents.

Another potential option to reduce the quantity of hydrogen in the containment is venting (filtered or unfiltered), which is possible only with systems not designed for the purpose. This option has not yet been analyzed. The existing measures for management of accidents during which the containment integrity is threatened by hydrogen are described in SAMG strategies that use all available means to prevent dangerous form of hydrogen burning.

Prevention of containment overpressurization

The design function of the containment is to prevent release of radioactive (Ra) substances into the environment or to limit radiation consequences of an accident in the surroundings. The containment forms the last barrier against activity release and it is independent of the other barriers. The function of the containment is ensured by its design and structure.

Dukovany NPP

The containment CTMT is sure to withstand the design overpressure 150 kPa and it can withstand a double of that value with a high probability. The containment tightness is checked regularly (within the PERIZ tightness test) and measures are implemented to increase the tightness. The design function of the containment is ensured with two methods:

- use of isolation fast acting valves on all routes that pass through the containment wall,
- use of tight passageways and tight penetrations for all pipings and cables that pass through the wall and minimization of leakage by limiting the duration of internal overpressure with a subsequent creation of underpressure in respect to the external environment.

The system that reduces pressure in the containment consists of two parts:

- The vacuum-bubbler condenser (VBC) system contains passive bubbler condenser trays that condensate water steam and subsequently at a higher pressure ensure passive spraying of the containment. Non-condensable gases and air from the containment are trapped in gas holders (traps) that are subsequently automatically isolated from the containment environment.
- Spraying system with three active spraying pumps.

The cooperation of both the systems guarantees generation of underpressure in the containment and complete elimination of release to the surrounding environment. A correct function of the bubbler condenser, which is important for the fulfillment of safety functions of containment at VVER 440/213, has been reviewed within the project PHARE/TATIONS PH2.13/95 "Experimental qualification of the bubbler condenser". Tests, experiments on the unique equipment modeling SG boxes and bubbler tower in the scale 1:100 and, finally, analyses have shown that the vacuum-bubbler systems for VVER 440/213 nuclear plants (Paks, Dukovany, Jaslovské Bohunice and Rovno) are able to resist loading and to maintain their function. The equipment is critical for limitation of the maximum pressure during accidents with large loss of coolant. It ensures the maximum contribution to pressure reduction until underpressure is achieved soon after the beginning of a large LOCA accident and thus prevents release of radioactive materials into the environment (for more details see chapter 2.3.3.4).

During the development of a severe accident it is not possible to continually maintain underpressure in the containment but results of the analyses indicate that it is possible to guarantee the minimum overpressure and the release of activity will be smaller than 0.1% of volatile fission products, excluding noble gases. In case of a hypothetic failure of active sprays the bubbler condenser will ensure a lower pressure in the containment than in a classical full-pressure containment and the release into the environment is lower than 1 % of volatile fission products, excluding noble gases. The vacuum bubbler condenser system thus eliminates the lower tightness of containment in comparison with full-pressure containments. This applies for a containment which keeps its integrity but after the loss of integrity it is necessary to anticipate a very high release of activity into the environment, which might be partly limited by a functioning active spraying system. The threat to the EDU VVER 440/213 containments posed by gas overpressure (with the exception of a short pressure increase during hydrogen burning) is very small. This is associated with the following facts:

- The vacuum bubbler condenser system condenses steam and creates under pressure in the containment at the beginning of the accident at the expense of certain over pressurization of its part – air traps.
- The total volume of the containment, including air traps, is in comparison with the remaining volume relatively big, around 50 000 m^3. The relatively high operating untightness of the containment represented by several percents of gas weight /day at the design pressure supports the pressure reduction. The untightness is probably due to tiny cracks in the concrete; however they may be sealed thanks to the effect of aerosols.
- The pressure of 250 kPa (overpressure 150 kPa) is the design pressure at which a major damage of the containment is still unlikely. Based on strength computations for the NPP, at the overpressure ca. 290 kPa the probability of a loss of containment integrity is ca. 5% and at the overpressure of 350 kPa the probability is 50%. Results of analyses of a potential loss of containment integrity caused by hydrogen overpressure indicate that after ca. 4.5 days, at the moment when debris penetrates the cavity wall, the overpressure in the containment would be ca. 120 kPa. If the cavity wall does not break it is estimated that the design overpressure would be achieved ca. after 5 days. Meanwhile, sealing of leaks would have a relatively big impact on the pressure development. Provided the leaks are not sealed the maximum overpressure reached after 4.5 days would be ca. 60 kPa. However, this scenario could be practically excluded because in case of a failure of heat removal there will be probably also the loss of water supply and interruption of steam generation.

The strategy of prevention of over pressurization is described in SAMG “Control of Pressure in the SG Box“ that is used already at the overpressure of 10 kPa. The purpose is rather to prevent a higher release through existing leakages than the future threat of containment over pressurization.

Temelín NPP

CTMT minimizes leakage to very low values even if the internal overpressure in the containment is high. The integrity of ETE containments is ensured according to the design by the following systems:

- system of containment isolation – isolation valves automatically close if the pressure in the containment increases; the operability is conditional on power supply;
- system for reduction of pressure in the containment – spray pumps and reserve tanks with chemical agents to trap post-accident iodine – the operability is conditional on power supply;
- system for liquidation of post-accident hydrogen – passive autocatalytic recombiners for design basis accidents – requires no electric power supply.

The design systems for pressure reduction include 3 divisions of the spray system and each of them is capable to reduce pressure in the containment by condensation of steam that leaks from a broken steam line or primary loop. Over pressurization of the containment during a severe accident might occur as a result of dynamic phenomena (i.e. burning hydrogen) or long-term accumulation of steam or non-condensable gases in the containment atmosphere.

The dynamic phenomena may lead to pressure peaks that may not be mitigated by normal heat removal from the containment (i.e. the increase of energy in the containment is bigger than removing capacity of spray systems). Analyses were performed to determine the limit case of pressure increase in the containment. The analyses have shown that until melting through of the reactor pressure vessel (RPV) and relocation of the melt to the bottom of the concrete reactor cavity the pressure in the containment cannot increase to values that may seriously threaten its integrity.

Only after the beginning of an interaction between the melt and concrete in the ex-vessel stage the pressure in the containment may further increase to values threatening its integrity. The strength computations for the containment indicate that after the design pressure in the containment is exceeded the containment structure initially demonstrates linear behavior. Subsequently, cracks in the concrete start developing on the inner side, the entire steel lining gradually becomes plasticized and finally the containment tightness is disrupted. The pressure which disrupts containment integrity (ca. 1.6 multiple of the design pressure 0.8 MPa, which corresponds to 5% probability of the damage) represents “cliff edge“ conditions from the viewpoint of threat to containment integrity due to overpressurization.

EOPs anticipate the use of CTMT spraying in a manner that the pressure inside CTMT remains within the design parameters. The operation of the spray system should in the long-term maintain the pressure in CTMT at a value which corresponds to the pressure of the surrounding atmosphere, unless there was a major release of non-condensable gases produced by reaction between the melt and the concrete. Measures for management of accidents that threaten containment integrity by high pressure are described in SAMG strategies that use all available means to reduce pressure in the containment. The respective strategies in SAMG provide instructions on how to perform preventive measures to reduce pressure in the containment provided its integrity is threatened by overpressurization. Ventilation of the containment with systems not anticipated for the purpose in the design has been identified as one of the potential activities for mitigation of serious threats to the containment by high pressure.

Prevention of re-criticality

Dukovany NPP

For VVER 440 the risk of boron dilution in an advanced stage of an accident is lower than for standard western PWR reactors. During the shutdown, due to the tandem control rods, fuel parts of the rods (37 out of 349) will be out from RC and the reactivity will be lower even in case of melting and relocation of the control rods. In case of a threat to core subcriticality in the preventive stage (EOPs) a higher concentration of boron is required particularly in order to compensate non-insertion of Emergency&Control rods and not to compensate the introduced positive reactivity from the decrease of temperature during cooling. After the fuel geometry is lost there is no problem relating to boron dilution. The geometry created by debris inside the reactor or in the reactor cavity is in all situations deeply subcritical, even in case of flooding with pure water.

Temelín NPP

In case that primary circuit is made up with water with a low content of H_3BO_3 criticality may be achieved repeatedly and the reactor power output may increase as a result of increased moderation of neutrons, as long as the original geometry of the reactor core is retained. Provided the original core geometry has changed and thus the ability of neutron moderation has been reduced then the criticality cannot occur.

A potential return of the reactor to power does not mean an immediate risk because it is limited by the formation of bubbles in RC. Following severe fuel damage and a loss of geometry of control clusters in RC a hollow may appear without any nuclear poisons. Due to the absence of moderator (under the given conditions the water always transforms into steam) the ability of moderation is lost. Due to the disrupted geometry of RC the criticality cannot occur in a big volume. Measures that prevent a decrease of boron concentration have the highest priority during activities according to EOPs in the preventive stage before the fuel damage, when the original geometry of RC is fully retained, which enables moderation of neutrons and development of criticality. When performing activities according to SAMG, in the stage after serious fuel damage and a loss of the original RC geometry, the respective strategies also describe measures for the accelerating power growth, nevertheless, due to the disrupted RC geometry, the criticality cannot occur in a big volume of the core.

2.2.3.4 Spent fuel storage pools (SFSP)

Dukovany NPP

The spent fuel storage pools are situated in the reactor hall shared by two units outside the containment. The spent fuel storage pools (SFSP) are cooled by two cooling circuits. Each cooling circuit includes a circulation pump and a heat exchanger. Heat exchangers are cooled by service water essential (ESW 1 and ESW 3). A deep subcriticality of the spent fuel in the storage pool is guaranteed both by the coolant with boron concentration 12 g/kg and by borated steel in the structure of storage racks. The use of borated steel alone guarantees subcriticality even in case that the spent fuel is cooled with pure water.

The issues of SFSP cooling or leakage of coolant from the cooling circuit are addressed in EOPs. After the interruption of heat removal from SFSP the temperature would continually increase, which would be important particularly in the case that the upper rack is full. If the heat removal is not recovered the upper layer of fuel would be uncovered first, with the subsequent risk of damage of fuel cladding and melting of fuel in an early stage of a severe accident. Because the storage pools are not situated in hermetically separable premises (they are protected only by the reactor building shell) this would be followed by release of radioactive substances into NPP surroundings. Hydrogen would be released into the reactor hall in case of steam-zircon reaction. Due to the existence of an alternative method of heat removal, by means of heat accumulation in ECCS tanks, a long-term loss of heat removal from SFSP is not expected because, from the viewpoint of time to perform activities for recovery of cooling of spent fuel stored in SFSP, the situation is more favorable than in the case of a loss of heat removal from RC. The accumulation capacity of completely filled ECCS tanks is ca. 4 days. No detailed procedures have been developed yet for this kind of actions. An alternative method has been considered to make up coolant from bubbler condenser trays, coolant from the adjoining unit and making-up of SFSP using LFRS means. However, no detailed procedures have been developed so far. When using coolant from all ECCS tanks and bubbler condenser trays the coolant supply will be sufficient for making-up of losses due to coolant boiling in SFSP for over 8 days, even in case the fuel is arranged in two racks one over the other. If none of the above-described methods can be used then the same alternative methods may be used for SFSP cooling as for SBO: making-up of SFSP from the higher located VBC trays by gravitation; an open reactor can be made up with coolant from hydro accumulators or SFSP may be filled with water using LFRS means. The current status of the design does not provide any alternative stable systems for cooling or making-up of coolant into SFSP. In case of an open reactor during refueling there is also the option to supply coolant with any pump of the ECCS high-pressure or low-pressure system, directly into the reactor interconnected with SFSP and from there to I.C.

A total loss of electric power supply (SBO) leads to a loss of forced heat removal from SFSP by means of ESW. In case of SBO on only one unit the power supply can be provided to SFSP cooling pumps system from the twin unit via a servicing line. The procedure is sufficiently described in EOPs. SBO does not pose a problem from the viewpoint of assurance of sufficient subcriticality. The geometry and material of the storage racks ensures sufficient subcriticality, even in case of coolant boiling or if SFSP is filled with water containing no H_3BO_3. A loss of electric power supply means a loss of availability of SFSP cooling systems. In case of SBO the forced heat removal from SFSP is immediately interrupted and the temperature gradually increases, which is important particularly when the upper grate is fully occupied. If the heat removal is not recovered the temperature would increase to the boiling point of the coolant in SFSP. The coolant in SFSP would start boiling out and if none of the below identified methods for making-up of the evaporated coolant is used the fuel would be exposed in an early stage of the accident.

This means that for SBO the design has no diverse system. However, the heat removal can be performed by alternative methods:

- for an open reactor the coolant may be supplied from hydro accumulators;
- for making-up of SFSP from the higher-located VBC trays which can be used to maintain the level in SFSP by means of gravitation flow from bubbler towers trays, the coolant supply to make up for the evaporated medium is ca. 13 days. This method of alternative making-up of SFSP is mentioned EOPs but specific procedures for interventions on the site have not been developed;
- another alternative method is the use of firefighting techniques for making-up of coolant and keeping the fuel temperature in SFSP. In this respect the pool is easily accessible for LFRS techniques (via a rail siding corridor). This extreme case of SFSP cooling uses water supplied into the reactor hall with mobile pumping techniques while the boiling coolant evaporates back into the reactor hall.

Emergency operating procedures (EOPs) describe the above-mentioned alternative methods to make up coolant into SFSP but specific procedures for interventions on the site have not been developed. All fuel placed in SFSP is proved as sealed and in case of untight fuel it is placed into hermetic cases in the SFSP racks and they function as a barrier against release of radioactive substances and ensure sufficient passive cooling of the fuel assembly. Therefore the use of alternative methods, to keep sufficient level in SFSP by making-up of coolant in order to prevent fuel damage, means that potential evaporation of the boiling coolant from SFSP into the reactor hall will not lead to significant release of Ra substances into the reactor hall.

Note: An analysis of the development of accidents in the storage pool for shutdown conditions has been planned for 2012. It will analyze behavior of the pool in the mode 6, i.e. during refueling, mode 7 during total removal of fuel from the reactor and during mode 1 through 5 in which the storage pool is jointly with the reactor hall hermetically separated from the containment.

Temelín NPP

VVER1000/320 units have SFSP situated inside the containments, immediately next to the reactor.

Even though there would have to be multiple failure of defence in-depth before the total loss of heat removal from spent fuel in SFSP, the consequences of such a condition would be so serious that additional measures have been proposed to improve the already very robust design from the viewpoint of assurance of heat removal from SFSP into the ultimate heat sink, either due to SBO or a loss of UHS.

The purpose of the measures is to strengthen the levels of defence in-depth after the current beyond-design basis initiation events (earthquake, floods, extreme conditions, results of human activities, etc.), which may result in a loss of ability to perform safety functions. Similar means as for heat removal from RC are used to ensure electric power supply to systems and heat removal into the ultimate heat sink from spent fuel in SFSP. Each section of SFSP with spent fuel is cooled with one cooling circuit. Each of the three cooling circuits includes a circulation pump and a heat exchanger. Heat exchangers are cooled by ESW. Pumps cooling SFSP and ESW pumps are also power supplied by DG SPSS of the safety systems.

When SFSP are operated in the fuel storage mode the level is required to be maintained at more than 792 cm. Provided the level in SFSP drops below 550 cm the heads of the stored fuel assemblies will be uncovered. In case of a total loss of normal SFSP cooling (either due to the low water level or interruption of heat removal) SFSP is made up with the containment spray system alternatively set up for emergency making-up of SFSP. Using this system for making up SFSP with water overflow ensures removal of coolant from SFSP to the containment bottom and subsequently into the containment sump, which ensures an alternative method for heat removal from the spent fuel in SFSP via ECCS cooler. This cooling circuit is independent of the system cooling of SFSP and provides an alternative method of heat removal from spent fuel. Heat from SFSP is also removed by evaporation into the containment and the evaporated quantity is compensated with the containment spray system.

If the heat from spent fuel in SFSP cannot be removed with ESW into the ultimate heat sink then this method of SFSP cooling is in the long-term limited with the passive thermal capacity of CTMT. Computations have been used to analyze the loss of cooling of SFSP with the stored spent fuel. The computations have shown maximum temperatures achieved in SFSP during storage with limited cooling, heat-up rates and time margins before achievement of the saturation temperature and before uncovering of heads of the stored fuel assemblies after a loss of SFSP cooling. Results of computed heat-up rates and time reserves before boiling are dependent on many factors, such as the number of fuel assemblies in individual SFSP sections (heat output), time after the removal of fuel assemblies from RC, level in SFSP at the moment of loss of heat removal, initial temperatures in SFSP, etc.

Based on the completed analyses it is possible to conclude that, depending on the initial conditions, the trend of temperature increase in SFSP after interruption of cooling ranges from several units to several tens of °C/h and the boiling margin ranges from several hours to several tens of hours. At the maximum thermal load of SFSP and after a loss of heat removal from SFSP the stored fuel assemblies will be damaged not sooner than in the late stage of the accident. In respect to the time to perform activities for recovery of cooling of spent fuel in SFSP, the situation is more favorable than in case of a loss of heat removal from RC; nevertheless, a long-term loss of heat removal, exceeding several tens of hours without an alternative method to make up water, might lead to a damage of the stored spent fuel in SFSP.

A loss of electric power supply would interrupt cooling of spent nuclear fuel and lead to heating of water in SFSP. The trend of temperature increase in SFSP after the interruption of cooling depends on the initial conditions (time since the removal of spent fuel from the reactor, quantity of fuel in SFSP, etc.). Even at the maximum heat load SFSP is not threatened by immediate damage of the stored spent fuel after a loss of heat removal from SFSP; the fuel may be damaged only after tens of hours after SBO. If the maximum residual power is generated in SFSP (fuel from the entire RC has been removed and the rest of SFSP is filled with spent fuel from previous campaigns) the minimum time to achieve saturation temperature is ca. 30 h. The volume of water in SFSP in the mode of fuel storage is in each of the sections 01 and 03 ca. 223 m^3 and in the section 02 ca. 104 m^3 (in the refueling mode it is approximately twice bigger). With regard to the above-mentioned volumes of the individual

SFSP sections, the coolant available in the containment sump will double the time to saturation temperature (ca. 60 hrs) and the available coolant in supply tanks for refueling will extend the time interval to saturation temperature four times (ca. 120 hrs). Another important aspect, which significantly affects the time before the saturation temperature is achieved in SFSP, is the water level. If the level in SFSP drops below 754 cm then a loss of circulation via the SFSP cooling system will occur and if the level in SFSP drops below 550 cm the heads of stored fuel assemblies will become uncovered. In case of a loss of the last functional redundancy of the heat removal system from SFSP and achievement of the saturation temperature it is necessary to remove the heat by boiling the coolant in SFSP and its evaporation into the containment.

The ESW system is used to cool down I.C into the cold condition, to remove heat from spent fuel in SFSP and from appliances of safety systems and systems related to nuclear safety; the system transfers heat via CBSS to the atmosphere which serves as the ultimate heat sink. All three ESW systems are in operation under standard conditions (redundancy 3 x 100%). Heat is removed from each ESW system into a separate CBSS where it is transferred to the atmosphere by evaporation of water from the water level and from water sprayed from nozzles.

In case of a loss of UHS/ESW the situation from the viewpoint of heat removal from SFSP is the same as during SBO, i.e. interruption of cooling of spent nuclear fuel and heating of water in SFSP, see above (potential damage occurs only after tens of hours in a late stage of the accident). In consideration of the several operating methods of heat removal and several alternative methods in case the former are not available, even in the highly improbable case of a loss of ability of ESW system to transfer heat from SFSP and safety systems equipment into the surrounding atmosphere it would be possible to find sufficient time for preparation of alternative methods for removal. However, the loss of ESW is always associated with inability to cool the unit into a cold condition and to keep it in the long-term in the cold condition. A different situation occurs in the outage mode (with an open reactor) when the heat removal from RC is dependent on ESW operation. A loss of ESW will cause an increase of temperature in RC. In this case it is possible to fill pools for wet transport. Without the heat removal the temperature in the wet transport pools will increase to the saturation temperature – see also above. If the evaporated quantity is compensated by making-up the heat in this mode may be removed in the long-term. From the long-term prospective it is necessary to resume operation of the ESW system at least in one safety division, which will make it possible to cool down the unit into the cold condition.

Limitation of radioactive release after a severe damage of spent fuel in the spent fuel pool

In an extreme case of a long-term loss of ability to cool SFSP or to remove heat into UHS the fuel in SFSP could be damaged within hours (ca. 12 hours at EDU and 30 hours at ETE)

Dukovany NPP

Procedures to address accidents associated with fuel melting in SFSP are not available yet. Neither MCR staff nor technical support center (TSC) staff have the so-called shutdown SAMG (SAMG for shutdown conditions), nevertheless the available options are known and they consist in continuing making-up of water and heat removal and potential isolation of leakage from SFSP according EOPs. The damage would occur after a relatively long time, with the exception of mode 7 (all fuel removed from the reactor into SFSP), which gives sufficient time for operative solution. The principal measure to limit release into the environment is to stop or slow down the accident by flooding SFSP with water. An emergency system for pool flooding is being prepared, which will be combined with other measures in the reactor hall that exclude presence of personnel. The reactor hall has a big volume which has a positive effect on dilution of fission products. Additional potential measures limiting release are as follows: In case of activity release from SFSP (or from the reactor in mode 6) it is necessary to switch off immediately the big capacity ventilation systems for the reactor hall; this procedure is provided in the existing EOP for shutdown conditions. Once all the personnel leave the reactor hall it is essential to close all passages into the reactor hall. For the unit in modes 6 or 7, i.e. during refueling or removal of all fuel from the reactor, when the containment is usually connected with the reactor hall with several passages, the containment ventilation systems shall be switched off, all persons shall leave the containment and all access routes into the containment unit in modes 6 or 7 shall be closed. The measures are necessitated by the fact that it is impossible to quickly separate containment from the reactor hall.

Temelín NPP

Technical means for mitigation of fuel damage in SFSP are available and the strategies consist in continuing making-up of water and heat removal and potential isolation of releases from SFSP according EOPs. The shutdown SAMG for accidents connected with fuel melting in SFSP have not been developed yet. No analyses have been performed for damage of spent fuel stored in SFSP. With regard to the method of making-up of SFSP with the containment spray system, the long-term loss of heat removal from SFSP is not anticipated without a concurrent loss of heat removal from RC. In case of a simultaneous loss of heat removal from SFSP and from RC (due to the location of SFSP in CTMT) the decisive measures are those resulting from the loss of heat removal from RC because, from the viewpoint of time to perform activities leading to recovery of cooling of spent fuel in SFSP, the situation is more favorable than the loss of heat removal from RC.

Instrumentation necessary for monitoring of the spent fuel condition and accident management

Dukovany NPP

Measurements characterizing the SFSP condition (temperature, water level, flow rate in the SFSP cooling system) are available on MCR panels in the computer network. Measurements of parameters connected with SFSP cooling are not available in emergency control room (ECR) or in PAMS but they are in the computer network. Similarly, PAMS does not have measurements about the Ra situation in the hall in the proximity of SFSP. Due to the big volume of the reactor hall, its lower tightness and low residual power of the fuel, the anticipated conditions are not as severe as inside the containment. Most measurements, however, will remain available. The most important of them are measurements of activity in the atmosphere and water level in SFSP.

Temelín NPP

The key parameter for evaluation of a loss of heat removal from spent fuel stored in SFSP is the water level in SFSP. As long as the fuel is covered with a layer of water (even when boiling) the residual heat will be removed from the fuel. Once the water boils out from SFSP and once the stored fuel assemblies are uncovered the fuel assemblies will start overheating. The water level in SFSP and several other parameters, such as the status of ESW systems and flow rate of ESW into the exchanger for SFSP cooling, are communicated via PAMS. Another parameter measured in SFSP is temperature.

In extreme cases, if the loss of ability of SFSP cooling or heat removal into UHS is not solved in the long-term, the fuel in RC and spent fuel in v SFSP could be damaged at both power plants due to the absence of alternative methods of heat removal from RC, SFSP and components cooled with ESW. At both power plants the trend of temperature increase in the most conservative case (for the least favorable initial conditions) would be several tens of °C/h and the margin to boiling would be several hours. The heads of the stored fuel assemblies could be uncovered within 12.6 h (EDU)/30 h (ETE) (limiting condition) with a subsequent potential cladding failure and fuel melting. At EDU this could be followed by a release of radioactive substances into the NPP proximity (the pools are not situated in hermetically separable premises) while at ETE the last barrier is the containment.

Another opportunity to improve resistance against SBO is the strengthening of the defence in-depth levels during initiation events beyond the existing design, e.g.:

- alternative means of AC power supply for the existing equipment that ensures cooling and heat removal from RC and SFSP, including the possibility of connection to the existing power supply distribution /technology,
- diverse means for cooling and heat removal from RC and SFSP, including the possibility of connection to the existing technology.

Another opportunity to improve resistance against UHS is the strengthening of the defence in-depth levels during initiation events beyond the existing design, e.g.:

- diverse means for cooling and heat removal from RC and SFSP, including the possibility of connection to the existing technology.
- use of alternative and diverse means – the so-called "Extensive Damage Mitigation Guideline" (EDMG) with the objective to ensure cooling and heat removal from RC and SFSP.

2.3 DESCRIPTION OF ACTIVITIES PERFORMED BY THE STATE REGULATORY AUTHORITY

2.3.1 Overview of implemented and planned activities

On 25 May 2011 SÚJB invited the Licensee to prepare reports about the “stress tests“, i.e. about the targeted analysis the purpose of which was to review safety and safety margins of both power plants in the Czech Republic – Dukovany and Temelín - with a focus on “Fukushima scenarios”. On 15 August 2011 the reports about the evaluation of both power plants, based on information collected from all relevant documents (safety reports, PSA studies, documents for regular safety reviews, rules for abnormal situations and accidents – procedures for extraordinary events, guidelines for management of severe accidents etc.) and based on personal inspections of important systems and equipment intended to verify their current condition, were handed over to SÚJB. During the evaluation SÚJB used sharing of experience with other operators of VVER reactors, as well as opinions provided by experts from the Nuclear Research Institute in Řež (ÚJV Řež a.s.) and SÚJB. SÚJB also requested a preliminary evaluation of both the reports by the Research Centre Rez.

2.3.2 Further steps to be taken by the state regulatory authority

The cooperation between SÚJB and the operator, as well as results of the stress tests and potential subsequent measures, have been described above.

SÚJB will be approving specific design changes resulting from the stress tests, as proposed by the operator.

In connection with comprehensive results of the stress tests SÚJB will also consider potential legislative adjustments/changes.

A schedule of the individual activities has not been completed yet.

2.3.3 Conclusions made by the state regulatory authority

Results of the review of the stress tests – targeted evaluation of safety margins and resistance of EDU and ETE against extreme natural conditions, loss of electric power supply, loss of heat removal into the ultimate heat sink and ability to manage the situation in case of scenarios leading to a severe accident for most accident scenarios – have confirmed safety and time margins and sufficiently robust barriers to ensure defence in-depth levels, both in terms of the design and personnel, administrative and technical provision of accident management (the high resistance of both power plants against extreme effects). No problem/condition was identified at either power plant that would require immediate measures. Both power plants are able to safely withstand even highly improbable extreme emergency conditions, without endangering the plant surroundings. Results of the stress tests have confirmed the fact that designs and actual state of both NPPs provide sufficient margins to avoid severe accidents.

In respect to external risks, the strengths of both power plants include in particular:

- robust and conservative design ready to face demanding conditions,
- design that is continually checked and reviewed against the current safety requirements,
- continual process of incorporation of new safety requirements,
- two big water reservoirs/dams for raw water at both power plants,
- a big supply of cooling water inside the power plants,
- compact racks of the SFSP ensuring subcriticality of fuel even in case of flooding with pure water,
- at EDU, a particularly big volume of hermetic premises (bubbler condenser system) and relatively smaller source term (lower reactor power parameters) and possibility to use diverse means for heat removal (fire pumps),
- at ETE the SFSP is located inside the full-pressure containment.

2.3.3.1 Evaluation of resistance of the power plants against a loss of electric power supply

The electric power supply sources at EDU and ETE ensure sufficiently robust design and level of safety assurance in case of an external loss of electric power supply. The design benefits from a high level of mutual independence between working and backup sources of on-site power and redundancy of secured power supply systems that supply safety-important systems and components and have their own emergency sources (DG and accumulator batteries). The unit operating on power has a higher design resistance against a loss of electric power supply than during outage for refueling. The least favorable case from the viewpoint of safety assurance is the loss of electric power supply at all/both units at a time.

On the EDU site there are 12 emergency AC sources (DGs) in total, while each DG has a supply of Diesel fuel for 6 to 7 days without the necessity to replenish fuel from external sources.

On the ETE site there are 8 emergency AC sources (3 emergency DGs for each unit and 2 DGs shared by both units) in total, while each DG has a supply of Diesel fuel for over 2 - 3 days without the necessity to replenish fuel from external sources (moreover, additional supplies of Diesel fuel are available on the site to extend the DG operation).

In the mode with a loss of external power supply the EDU and ETE units may be maintained in a safe condition in the long-term or cooled down to the cold condition or safety maintained in the outage mode (power supply is ensured for all necessary engineering systems and I&C systems) as long as at least one of the DGs is started for each unit.

In case of the total loss of AC power supply (SBO) safety systems and safety-related systems still have emergency DC sources of uninterrupted power supply (accumulator batteries). Without the operation of a respective DG the accumulator batteries are not recharged and their discharge time is in hours (ETE) or tens of hours (EDU), depending on the current load. This time is sufficient to recover on-site power supply from the nearby hydroelectric power plants Dalešice or Vranov (EDU) or from the hydroelectric power plant Lipno (ETE). The discharge time may be significantly extended by a controlled load relief of accumulator batteries by gradual use of individual divisions and use of high-capacity accumulator batteries of safety-related systems.

At ETE another alternative method of long-term charging of accumulator batteries could be the other AC sources available on the site. (This has been proposed as a measure to further increase the plant´s resistance against a loss electric power supply.)

2.3.3.2 Evaluation of resistance of power plants against a loss heat removal into the ultimate heat sink

The ultimate heat sink (UHS) for EDU and ETE units is the surrounding atmosphere. Unused heat from operation of a unit on power or residual heat after the reactor shutdown may be removed into the ultimate heat sink – the atmosphere – by several methods. The transfer of heat between safety-important heat sources and the atmosphere is ensured by the ESW system.

EDU has a water supply for ca. 39 days of operation of the ESW system to remove residual heat from shutdown EDU reactors without replenishment of water into the ESW system from external sources. There are 12 ESW pumps in total for one main production building (2 reactor units). A loss of all ESW pumps might occur in case of a concurrent loss of electric power supply at both units of the given main production building. Robustness of EDU in case of a potential loss of all ESW pumps corresponds to the scenario after SBO. If a loss of the ESW system is not combined with SBO then it is possible to use an alternative method of heat accumulation from SFSP into the ECCS system tanks or making-up of evaporated coolant from SFSP from bubbler condenser water trays. The accumulation ability of the full ECCS tanks is ca. 4 days and the supply of bubbler condenser trays to make up for the evaporated coolant is ca. 13 days. An alternative option is the use of firefighting techniques to make up for evaporated coolant and to maintain the temperature of fuel in SFSP.

At ETE the water supply in CBSS is sufficient for ca. 30 days of operation of the ESW system for removal of residual heat from shutdown reactors without making up of water into the ESW system from external sources. There are 6 ESW pumps in total per one unit. With regard to the spatial separation of the systems and pumps, the independence of electric power supply and other supporting systems, the simultaneous unavailability of all ESW pumps is extremely improbable. Even in case of operation of only one pump in one division of the ESW system, it is possible to ensure fulfillment of the basic safety functions.

2.3.3.3 Other potential safety improvements

Dukovany NPP:

- increase capacity of the system for liquidation of post-accidental hydrogen,
- develop “shutdown SAMG“ for outage /severe accident in SFSP.

Temelín NPP:

- alternative supply of Diesel fuel from a tank to ensure long-term operation of DG,
- alternative making-up of water into the containment sump,
- implementation of the system for hydrogen liquidation in the containment for severe accidents,
- verification of functions of equipment in beyond-design operating conditions,
- develop “shutdown SAMG“ (fuel damage in case of open reactor in SFSP).

2.3.3.4 Specific features of the VVER 440/213 (EDU) reactor containment

VVER 440/213 reactors operated at Dukovany NPP feature a specific design of the containment with a passive condensation system (bubbler condenser system), whose basic function is to reduce pressure in the mixture of air and steam in the hermetic zone of the reactor following the maximum design basis accident (LOCA) (guillotine rupture of the primary piping with the diameter of 500 mm) by condensation of water steam in special trays filled with a solution of H_3BO_3, with a subsequent isolation of non-condensed gases in hermetic air traps with check valves. By creating underpressure against the surrounding atmosphere the system at the same time minimizes potential releases of radioactivity outside hermetic premises. The system has been designed to keep its integrity under pressure and temperature conditions that occur in the hermetic zone after the maximum design basis accident.

The thermodynamic principle, on which the function of the bubbler system is based, is identical with the function of containment of western boiling water reactors (BWR). Due to limited information about the experimental verification of the system from authors of the original design and due to the need to expand knowledge and ability to model integral behavior of the system and partial physical phenomena in the conditions of large and small LOCA accidents, numerous international projects and studies were organized in 1990s; the parties involved included countries operating those types of reactors and also prominent western institutions, such as SIEMENS/KWU, EdF, Empresarios Agrupados, GRS, IRSN, etc. The first series of studies appeared within the so-called Extrabudgetary Programme organized by the International Atomic Energy Agency:

- Ranking of Safety Issues for WWER 440 Model 213 Nuclear Power Plants IAEA-Report WWER-SC-108 1995-02-21
- Strength Analysis of the Bubbler Condenser Structure of WWER 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 cardinal verification of functionality and structural strength properties of the bubbler condenser system was performed within the programs PHARE and TACIS organized by the European Commission. In order to verify the integral behavior of the bubbler system and to obtain credible experimental data suitable for validation of computation programs, an experimental stand was built in the research center EREC in Elektrogorsk in the Russian Federation within the PHARE/TACIS PH2.13/95 program “Bubbler Condenser Experimental Qualification - BCEQ“. The experimental equipment, jointly with smaller models of parts of the bubbler condenser system in the research institutes VUEZ Tlmače (Slovakia) and SVUSS Běchovice (Czech Republic), enabled the following:

- performance of experiments simulating an integral thermohydraulic and hydrodynamic interaction of steam with the bubbler condenser structure (under the management of Siemens/KWU),
- performance of static tests to verify strength properties of the bubbler condenser structure in VUEZ Tlmače (under the management of Empresarios Agrupados),
- performance of small-scale experiments to study partial thermohydraulic phenomena and to verify instrumentation in VUSS Běchovice (under the management of Electricité de France).

Experimental works within the project were accompanied with parallel analytical studies with the involvement of the research institute VEIKI, Budapest (Hungary). Results of the project have been summarized into a series of research reports, including:

- 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 has provided a positive, factual and objective proof that the bubbler system is qualified for the conditions of the maximum design accident of VVER 440/213 reactors and that it performs its safety functions under those conditions – it maintains integrity of the reactor hermetic zone and prevents release of radioactive substances into the environment.

The BCEQ project was not the last one organized for verification of functionality of the bubbler system. In 2001-2002 the international committee CSNI OECD/NEA sponsored a project called "Answers to Remaining Questions on Bubbler-Condenser", to answer some questions concerning conservative character of results of the BCEQ project, experiment scales, non-homogeneities in the flow and temperature field in the volume of bubbler condenser and, particularly, verification of functionality of the bubbler condenser system under conditions of a long-lasting small LOCA accident.

The project was initiated by regulatory authorities of the Czech Republic, Slovakia and Hungary and funded by utility companies of those countries. A management team of the project consisted of one representative from each of the above.-mentioned regulatory authorities and experts from German GRS, French IRSN, US DOE and EU. The project also included additional three experiments on the experimental stand in EREC, specifically:

- rupture of the main steam line,
- medium LOCA (rupture of piping with the diameter 200 mm),
- small LOCA (rupture of piping with the diameter 90 mm).

Conclusions from the tests and the final position of the Management Board on the outlined questions are summarized in the report "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 Management Board concluded the project with a statement that additional experiments have demonstrated that the loads to which the bubbler condenser system is exposed under the conditions of design accidents do not threaten integrity of the bubbler condenser system. This conclusion of the project was also accepted by the committee CSNI OECD Nuclear Energy Agency (OECD/NEA).

Conclusions of the stress tests confirm that the bubbler system may, apart from its classical safety function in the conditions of design basis accidents, play a significant role in a severe accident because the volume and quantity of water with the shutdown concentration of boric acid increase the supply of coolant that can be used for management of beyond-design basis accidents and, last but not least, the area of internal building and technological structures significantly limits the containment damage by overpressure and significantly reduces the potential release of radioactive substances outside the reactor hermetic zone.

2.3.3.5 Summary

Despite the fairly robust barriers, based on the results of evaluation of safety margins for initiation events, loss of safety functions and measures for management of beyond-design basis and severe accidents at EDU and ETE, it is possible to conclude that for the highly improbable beyond-design basis situations some opportunities have been identified for further improvement of safety/resistance of the power plant.

Each identified opportunity was classified from the viewpoint of importance for the size of safety margin, i.e. resistance against a potential loss of ability to perform basic safety functions and preparedness to manage the resulting situation. When assessing significance of the risks the number of defence in-depth levels was taken into account, which would have to fail before the occurrence of the given situation and the time for which the unit is able to resist with the existing safety margins. Until then it is necessary to have sufficient means to ensure the required functions or to adopt subsequent protective measures to limit irradiation and to protect persons.

2.4 FINAL SUMMARY OF CHAPTER 2

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 2.2.1) **Activity** **- Taken?** **- Ongoing?** **- Planned?**	(Item 2.2.2) **Schedule Or Milestones for Planned Activities**	(Item 2.2.3) **Results Available** **- Yes?** **- No?**	(Item 2.3.1) **Activity** **- Taken?** **- Ongoing?** **- Planned?**	(Item 2.3.2) **Schedule Or Milestones for Planned Activities**	(Item 2.3.3) **Conclusion Available** **- Yes?** **- No?**
Topic 2 Design Basis						
Stress tests of both nuclear power plants (EDU and ETE)	Completed on 31.10.2011	NA (For deadlines of the corrective measures, or implementation of the improvement - see below)	Yes	Ongoing (national evaluation completed, evaluation at the EU level has been under way.)	Results of evaluation by EU in presence of all regulators will be published in May 2012	Yes
EDU and ETE: Increase of the capacity of the system for liquidation of post-accident hydrogen	Planned	Medium-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
EDU and ETE: analyze the options to ensure shift personnel during events at several units	Planned	Short-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
EDU and ETE: analyze discharge times of accumulator batteries while applying a controlled relief of the load	Currently under way at EDU	At ETE planned within a short time horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
EDU and ETE: diverse means to make up water and to remove heat from SG, RC and SFSP (for more details see below)	Planned	Medium-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	Ne
EDU and ETE:	Planned	Short-term	No	Planned	Throughout	No

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 2.2.1) **Activity** - Taken? - Ongoing? - Planned?	**(Item 2.2.2)** **Schedule Or Milestones for Planned Activities**	**(Item 2.2.3)** **Results Available** - Yes? - No?	**(Item 2.3.1)** **Activity** - Taken? - Ongoing? - Planned?	**(Item 2.3.2)** **Schedule Or Milestones for Planned Activities**	**(Item 2.3.3)** **Conclusion Available** - Yes? - No?
procedure for renewal of power supply after SBO of all units on the site		horizon		Regulatory activity	the entire Licensee´s process	
EDU – making-up of water into SG by an alternative manner from demineralized water tanks 1MPa or from an external source if more than one unit is affected	Taken	Medium-term horizon	Yes	Planned Regulatory activity	Throughout the entire Licensee´s process	No
EDU – procedure for making up of SG of all four units with firefighting techniques	Ongoing	Short-term horizon	Yes	Planned Regulatory activity	Throughout the entire Licensee´s process	No
EDU – making-up of water into I.C/SFSP and cooling by an alternative method	Planned	Medium-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
EDU – ensure additional power supply for SPSS category I and selected appliances with SP category II (emergency shelters, telephone switchboards, TSFO etc.)	Planned	Medium-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
EDU – specification of procedures about filling the open reactor and SFSP by alternative methods	Ongoing	Short-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
EDU – develop procedures for loss of UHS and ESW systems at all units	Planned	Short-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 2.2.1) **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 2.2.2)** **Schedule Or Milestones for Planned Activities**	**(Item 2.2.3)** **Results Available** **- Yes?** **- No?**	**(Item 2.3.1)** **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 2.3.2)** **Schedule Or Milestones for Planned Activities**	**(Item 2.3.3)** **Conclusion Available** **- Yes?** **- No?**
EDU – implement measures to ensure a diverse means for UHS to CT	Ongoing based on PSR findings	Medium-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
ETE – ensure alternative making-up of water into SG/SFSP/I.C in case of open I.C	Planned	Medium-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
ETE – ensure alternative power supply for recharging of accumulator batteries and selected appliances	Planned	Medium-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
ETE - alternative supply of Diesel fuel from a tank for long-term operation of DG	Planned	Short-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
ETE – implement reconnection of valves that insulate containment of ventilation systems to accumulator batteries	Planned	Medium-term horizon	Ne	Planned Regulatory activity	Throughout the entire Licensee´s process	Ne
ETE – develop procedures for potential use of emergency DG of the adjoining unit in case of SBO	Planned	Short-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
ETE – develop procedures for insulation of the containment in shut down conditions	Taken		Yes	Planned Regulatory activity	Throughout the entire Licensee´s process	No
ETE – analyze the option to remove heat from SFSP without making-up	Planned	Short-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 2.2.1) **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 2.2.2)** **Schedule Or Milestones for Planned Activities**	**(Item 2.2.3)** **Results Available** **- Yes?** **- No?**	**(Item 2.3.1)** **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 2.3.2)** **Schedule Or Milestones for Planned Activities**	**(Item 2.3.3)** **Conclusion Available** **- Yes?** **- No?**
ETE – develop procedures for operation of units during long-term power supply from emergency sources	Planned	Short-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
ETE – alternative making-up of water into the containment sump	Planned based on PSR	Medium-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No
ETE – alternative sources and means of communication	Planned	Short-term horizon	No	Planned Regulatory activity	Throughout the entire Licensee´s process	No

3. MANAGEMENT OF SEVERE ACCIDENTS AND RECOVERY OF SAFETY FUNCTIONS OF UNITS ON THE SITE

3.1 INTRODUCTION

Considering of aspects of beyond-design basis and severe accidents has been for a fairly long time one of the main development trends in the efforts to improve safety of nuclear power plants (NPPs) in countries with developed nuclear energy industry.

Questions about how to include severe accidents into designs and operation of the existing NPPs are also addressed by the working group for harmonization of nuclear safety of NPPs in WENRA countries, with pro-active involvement of SÚJB. The WENRA report formulates the minimum reference requirements (reference levels) for the level of legislation, regulatory practice and condition of the operated NPPs for selected safety topics. Two topics of the overall number of 18 concern the requirements for beyond-design basis and severe accidents:

- Safety issue F: improvement of the designs of existing NPPs, including formulation of requirements for the selection and analyses of beyond-design basis accidents, for instrumentation usable in conditions of beyond-design basis accidents and for technical measures that ensure containment integrity during selected accidents,
- Safety issue LM: introduction of emergency operating procedures (EOP) and severe accidents management guidelines (SAMG), including formulation of requirements for their scope, form and content, verification and validation, confirmation and updating and for respective training of the personnel.

3.1.1 Legal environment

This area is regulated by the SÚJB Decree No. 195/1999 Coll., which establishes safety objectives and principles and requirements for nuclear installations with reactors.

SÚJB has also issued the guide “On requirement for nuclear installation design BN-JB-1.0.“, which, in conformity with the safety guideline IAEA NS-G-2.15, includes events of “extended design conditions“ and declares specific requirements also for BDBA.

In respect to beyond-design basis accidents the guide contains, among other things, also the following provisions:

Evaluation of Safety

(36) A combination of deterministic and probabilistic methods and engineering judgment shall be used to select the beyond-design basis events that are most important from the safety point of view (the so-called extended conditions), to perform their safety analyses and to identify events for which is it necessary and at the same time reasonably practicable to perform appropriate preventive or mitigating technical and organizational measures in the nuclear installation design.

(37) For analyses of these beyond-design basis accidents it is possible to use less conservative acceptance criteria and realistic assumptions – the so-called “Best Estimate” approach (the single failure criterion does not have to be applied, it is possible to consider interventions of systems classified as non-safety, etc.).

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

- to identify practicable measures to prevent occurrence and development of the accidents and to manage and to mitigate their consequences,
- as a basis for development of manuals for management of the accidents and for personnel training ,
- as a basis for development of plans for protection of the personnel and the population and for implementation of mitigating measures to reduce the impact of radioactive releases on the personnel, population and the environment,

Reactor Pressure and Cooling Circuit

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

Containment System

(108) Design criteria (including limits for temperatures and pressures inside the containment system and its tightness) must be specified to protect the containment system and to ensure its function and the design must guarantee that these criteria are complied with:

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

(117) The design must guarantee that the loss of the safety functions of the containment is virtually impossible and procedures must be specified and technical means and organizational measures shall be ensured for maximum protection of the integrity and functionality during beyond-design basis accidents, including severe accidents, in order to minimize consequences of potential overpressurization, overheating, damage by explosive gases or melt from degraded remains of the core, release of radioactive substances in liquid form and aerosols, core melt, etc.

Severe Accidents Management Guidelines

The severe accidents management guidelines – SAMG – were introduced for the first time in the Czech Republic within the commissioning of Temelín NPP. Experience of the Westinghouse company was used for their development.

SÚJB requirements for management of accidents are currently summarized in the SÚJB guide “Requirements for introduction of operating procedures of EOP and SAMG type“, BN – JB – 1.11. The guide specifies requirements for an accident management program, including operating procedures to be followed during management of design basis and beyond-design basis accidents, including severe accidents. The guide contains requirements for the format, scope and content of the procedures, including their maintenance and training of personnel. Most of the requirements in the guide are based on the IAEA safety standard – Management of severe accidents NS-G-2.15. The requirement to introduce operating instructions of EOP and SAMG types is based on the following provisions of the guide BN – JB – 1.11:

(3.17) The personnel carrying out measures required by accident management must have suitable operating instructions in form of a procedure or manual.

(3.45) The development of the accident management program shall be carried out in the following steps:

- Identification of vulnerabilities (weaknesses) 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
- Identification of the NPP's capabilities (potential) to resist challenges to critical safety functions and barriers against release of fission product, including capabilities to mitigate such challenges, in terms of both equipment and personnel.
- Development of suitable accident management strategies and measures, including hardware features to cope with the identified vulnerabilities
- Development of procedures and guidelines for accident management.

(3.25) Manuals for severe accident management shall take into account specific threats relating to reactor shutdown modes and long-term NPP outages involving open containment. The manuals shall include potential damage of irradiated nuclear fuel, both in the reactor vessel and in the spent fuel storage pools. Because general maintenance is carried out during planned NPP outages the manuals shall primarily focus on personnel safety.

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

In the SÚJB regulatory practice the above-mentioned requirements are transformed into binding conditions of resolutions issued by SÚJB regarding the license to operate the plant. For example, the license to operate the Dukovany NPP units issued in 2005 and 2007 contained the following condition:

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

Similarly, licenses from 2004 and 2005 for operation of units 1 and 2 of Temelín NPP contain the following condition:
"The Licensee shall update the severe accidents management guide (SAMG), including instructions for the unit's Control Room and Technical Support Centre. SÚJB shall be informed of all updates once a year by the end of the 1st quarter of the following year."

These conditions have been complied with by both power plants.

Information on Periodic Safety Reviews (PSR) and Probabilistic Safety Assessments (PSA) are provided in subchapter 1.1.2.

All these analyses deal with the resistance of units against occurrence and development of severe accidents and enable to look for other potential solutions to reduce risks of their consequences.

The ongoing evaluation only summarizes and complements the analyses completed by now and evaluates sufficiency of the adopted measures. Strategies are proposed to address any identified weaknesses.

3.2 DESCRIPTION OF ACTIVITIES PERFORMED BY THE LICENSEE

3.2.1 Overview of implemented and planned activities

This chapter focuses on measures to be adopted for mitigation of consequences in case of a severe damage of the reactor or spent fuel storage pool in order to prevent an extensive release of radioactivity.

Effective implementation of the severe accident management program and the plan for internal renewal represents comprehensive activities. The implementation requires substantial personnel sources, development of scenarios and knowledge of results of severe accident analyses, development and validation of procedures, availability of equipment and extensive training. The accident at NPP Fukushima Daiichi has demonstrated that severe accident sequences may be significantly more complicated during disastrous external events because of unavailability of fundamental equipment and radioactive leakage, operation of several units on the site, extensive loss of power supply, communication failures, extensive site damage or for other reasons.

The report contains results of evaluation of severe accident management and activities of internal renewal.

3.2.1.1 Approach to severe accident management and implementation of measures at NPP

The system of severe accident management at ČEZ nuclear power plants is provided for with a set of measures of personnel management, administrative and technical nature. Apart from this chapter 3, more information is provided in chapter 5, e.g. information about organization of emergency response by the Licensee in the subchapter 5.1.2.

Organization and strategy of severe accident management at NPP

The strategy of management of abnormal and emergency conditions is based on a logical development of any event at NPP. Intervention procedures for employees or other persons in selected working positions included in the emergency response organization (OER) have been developed for cases of extraordinary events in order to manage and to perform applicable interventions - for more information see chapter 5.

The basic objective of NPP safety is to prevent uncontrolled release of radioactive substances, particularly those generated in the reactor core. In order to meet the objective the design is based on the concept of the so-called defence in-depth, which consists in utilization of multiple physical barriers to prevent release of radioactive materials. The objective of severe accident management is to provide for level 4 of the defence in-depth (mitigation of consequences of a severe accident), following a failure of level 3 of the defence in-depth (i.e. failure of prevention of fuel damage during management of design and beyond-design basis events).

Implementation of interventions at NPP during extraordinary events is provided for in the first stage of development of the extraordinary event always by the personnel in continuous shift operation (IOER – internal emergency response organization of) managed by a shift engineer (SE).

The announcement of an extraordinary event is fully in competence of SE and the procedure is provided in chapter 5. Strategies to deal with technological accidents (until the fuel damage) have been developed and provided in the emergency operating procedures (EOPs). Strategies to mitigate consequences of accidents after fuel damage (severe accidents) have been developed and provided in the severe accident management guidelines (SAMG). The main priority of EOP is always recovery of heat removal from RC and prevention of damage of the 1st barrier against release of fission products (fuel cladding), while the main priority of SAMG is to prevent damage of the 3rd barrier against release of fission products (containment), which is the last intact barrier at that time.

SAMG developed for this stage of the accident describe activities to achieve a controlled stable condition.

The following objectives shall be met to mitigate consequences of a severe accident:

Primary SAMG objectives:

- to renew heat removal from RC or from the melt = to resume a controlled and stable condition of the heat generating source
- to maintain containment integrity as the last barrier against release of Ra substances into the surrounding environment = to ensure controlled condition of the containment,
- to stop release of Ra substances into the surrounding environment.

Secondary SAMG objectives:

- to minimize release of Ra substances into the surrounding environment in the course of activities to meet the primary objectives,
- to ensure the maximum availability of the equipment in the course of activities to meet the primary objectives.

A symptom-oriented approach is used consistently in management of emergency conditions, including severe accidents. The basic principle of the approach is that the corresponding strategy is selected based on the current development of the accident, as identified from positive symptoms (characteristics). Provided the symptoms change during the accident management and the selected strategy can be no more applied then the structure of the procedures and guides makes it possible to change the original strategy and to continue with activities described in another procedure or guide which is more suitable for the newly arising conditions. Continual diagnosing of the unit condition in the course of an accident therefore enables to correctly respond to potential changing conditions of accident development and the interventions always represent an optimized response to the given condition of the unit, taking into account also external events and imminent risks.

Procedures, training and practicing

The concept of management of technological accidents is based on the symptom-oriented approach.

The following strategies are currently being developed for Dukovany NPP and Temelín NPP for management of beyond-design basis and severe accidents:

- symptom-oriented emergency procedures for on-power conditions (EOP)
- symptom-oriented emergency procedures for shutdown conditions, including cases of threatened heat removal from spent fuel stored in SFSP (shutdown EOP)
- instructions for decision-making by TSC
- severe accident management guide for on-power conditions (SAMG).

All the above-mentioned procedures and guides have been developed and updated in cooperation with the Westinghouse Company.

A basic precondition for the performance of activities under the emergency procedures (EOP) is a reactor core condition which enables its cooling, i.e. RC is in a geometric condition which allows to cool it. The philosophy of EOP procedures includes continual assessment of the condition of physical barriers against activity release by evaluation of critical safety functions. The assessment ensures a timely identification of a worsening safety condition of the unit and it guarantees that a timely correction in case that a negative trend is identified during the development of an event. The purpose of EOP is to achieve and to maintain a safe condition of the unit in the long-term. However, in case of an irreversible damage of the reactor core the emergency procedures no more provide the optimum instructions to deal with the emergency situation. EOP can be no more used under those conditions and it is necessary to start activities under SAMG. At that moment also the main priorities will change. In case that the event passes into a severe accident the procedure focuses on maintaining the integrity of the remaining barrier against release of radioactivity, i.e. the containment.

The entire process of development and implementation of EOPs and SAMG is based on the symptom-oriented approach to unit control in emergency situations, which has been developed within the Westinghouse Owners Group for units delivered by Westinghouse to power plants in the United States and elsewhere, and its application to the VVER design. Also a well-tested approach has been taken over for verification, validation, implementation and training.

EOP, SDEOP and SAMG are regularly updated, including findings both from practicing of their use on the simulator or during emergency drills. External findings (within the „users group“ and the long-term cooperation with the Westinghouse company) are reflected in the documents through the so-called “maintenance program“.

The development of an emergency situation, apart from the type of the operating documents used during activities in response to a given situation, is closely connected with activities of the emergency response organization in agreement with the internal emergency plan (declaration of the level of an extraordinary event).

Preparedness of the shift and technical personnel to manage technological accidents is regularly verified during training on the full-scope simulator in presence of technical support center (TSC) personnel and during emergency drills. Emergency drills are conducted at least 4 times a year so that each shift of the standby organization of emergency response (SOER) participates in the drill at least once a year. The drills include also preparations for variants of operative interventions under aggravated conditions. Appropriate procedures have been developed for activities of emergency teams under aggravated conditions for their protection. The actual drill in the use of SAMG (under the supervision of Westinghouse experts) during management of severe accidents at Dukovany NPP and Temelín NPP was conducted after the introduction of SAMG into use.

Possibilities of the existing equipment

This method anticipates that standard means may be used beyond the framework of their purpose foreseen in the design, e.g. during beyond-design basis accidents. Even the so-called non-standard solutions may be used to remove heat from RC, e.g. by simultaneous cooperation of operating and safety systems.

Additional strategies have been developed for a beyond-design, highly improbable situation, with a total loss of ability to remove heat from the reactor core (loss of secondary heat removal simultaneously with a loss of ability of primary feed & bleed), in order to ensure secondary heat removal with the utilization of the existing equipment beyond the framework of their designed purpose.

Possibilities to use mobile equipment

The company fire rescue services (LFRS) are available at both Dukovany NPP and Temelín NPP sites with adequate firefighting techniques and they have been trained to intervene at any place on the site.

EOPs and SAMG for Dukovany NPP already include the use of mobile LFRS means on the site. Mobile LFRS pumps can be easily used as an alternative method for making-up of demineralized water directly into SG. Another alternative option is the use of firefighting techniques to make up for evaporated coolant and to maintain temperature of fuel in SFSP.

According to the applicable legislation, it is also possible to deploy basic and other components of the integrated rescue system (company medical center, Czech Police, Czech Army…). Based on the level of extraordinary event on the nuclear facility the individual groups operating within the rescue system perform tasks leading to liquidation of the extraordinary event on the affected facility or to limitation of its consequences.

3.2.1.2 Measures for assurance and management of power supply

Dukovany NPP

A key role in the assurance of long-term electric power supply at Dukovany NPP is played by the power supply from Diesel generators. The supply of Diesel fuel in the operating tanks for each DG is available for at least 6 hours. Each DG has also one mutually connected pair of supply tanks, with the minimum volume of 110 m^3 of fuel. The Diesel fuel is re-pumped from the supply tanks into the operating tank automatically, based on the level decrease in the operating tank. The pumps for fuel delivery are power supplied from the respective DG. The overall supply of Diesel fuel is 114.5 m^3 and it is sufficient for operation of one DG for at least 144 hours (in reality ca. 160 h), i.e. 6 to 7 days without the necessity to make up the fuel from external sources.

Demineralized water is made up into SG from the existing supply of demineralized water in the tanks 3x 1000 m^3 for each double- unit, which is sufficient for 72 h for all 4 units. Jointly with the supply of coolant in the feed water tank (FWT), the coolant available for making-up of SG of all four units of the plant is sufficient for ca. 4 days. Apart from the coolant in the tanks of demineralized water, steam generators may be alternatively also supplied with mobile means from cooling tower pools or other sources.

When using a conservative approach, the water systems at Dukovany NPP (when considering only halves of central pumping stations CPS I and CPS II, the level in the cooling towers at the minimum of -2.55 m) provide ca. 75 564 m^3 of water. This supply is sufficient for 931 h (ca. 39 days) of residual heat removal (i.e. operation of ESW pumps) after the reactors shutdown without making-up of water into the Dukovany NPP systems.

Temelín NPP

Diesel fuel management system is available on the site as a supply for long-term operation of DG and it can be used also for other potential mobile Diesel aggregates. All design Diesel generators available on the site have their own Diesel fuel tanks, which have been sized for autonomous operation (without making-up of Diesel fuel) at the maximum load:

- for emergency DG for at least 48 hours (in reality the time is even longer),
- for the shared DG (supply of appliances from both units) for ca. 12 hours.

In case a longer operation is required it is possible to get more Diesel fuel through a connecting piping on the technological bridges from the Diesel fuel management system.

Nevertheless, pumps in the Diesel fuel management system are not supplied from the safe (secured) power supply and therefore more Diesel fuel would be delivered by tank trucks.

Demineralized water is made up into SG from supply tanks 3 x 500 m^3 of the emergency SG supply for each unit and also tanks 2 x 770 m^3 shared by both units. This supply is sufficient with a margin to cool down the units into a cold condition (based on the design the emergency supply system for SG alone is sufficient to cool down the unit into the cold condition) or to maintain the units in a hot condition for ca. 72 hours.

Due to the existence of three redundant ESW systems it has been demonstrated that the heat removal into the ultimate heat sink can be ensured for at least 30 days, subject to the condition that all safety divisions will be used gradually or that the supply from CBSS of non-operable ESW systems will be re-pumped with mobile means to the operable ESW system.

3.2.2 Further steps to be taken by the Licensee

3.2.2.1 Measures available for increase of capacities for accident management

Despite several diverse systems for implementation of each strategy for accident management, opportunities for further safety improvement have been identified also in the personnel ability to manage severe accidents.

In the administrative management area it is particularly the severe accident management guide for shutdown conditions (shutdown SAMG), which has not been finalized yet. Nevertheless, EOPs, SDEOPs and SAMG are regularly updated by means of the so-called “maintenance program“.

In respect to personnel, there may be problems with their availability on the site, or usability of the emergency control center (ECC) and thus with the management of activities, with decision-making about high-risk variants of solution during management of emergency situations and, last but not least, with communication and warning of the personnel.

Plans for further improvement of resistance of the existing systems seek to evaluate preparedness for management of extraordinary situations from the backup emergency centers (in case the site is not accessible) and in a periodic review of nomination of the professionally best SOER personnel.

In order to improve effectiveness of the accident management system measures will be further elaborated in the following areas:

- organizational provision of the maximum effective use of the existing capacities or definition of additional capacities for management of predictable NPP conditions (all the site is affected, loss of control centers of emergency preparedness, loss of systems for communication and warning, decision-making about risky variants of solution, personnel alternation, extreme natural conditions…)
- finalization of some technological procedures/instructions/manuals for management of selected beyond-design basis conditions and severe accidents at NPP (shutdown SAMG, SAMG for damaged fuel in SFSP, EDMG, …) with the objective to ensure cooling and heat removal from RC and SFSP and to prevent radioactive releases
- improvement of personnel training in severe accident management (use of a simulation tool for displaying of parameters, phenomena and behavior of the unit for specific scenarios of severe accidents)
- additional technical measures to ensure non-technological supporting functions (access to the objects, availability of firefighting techniques, provision of ECC and shelters, systems of physical protection, …).

- alternative means to ensure long-term functional communication between all groups of the accident management system.

3.2.3 Conclusions made by the Licensee

3.2.3.1 Performance of NPP stress tests

Stress tests of nuclear power plants requested by the European Council are defined as a targeted evaluation of NPP safety margins and resistance in response to the events in Japan at NPP Fukushima Daiichi after the earthquake and subsequent tsunami on 11 March 2011. The task specification required an analysis of a combination of extreme situations that may lead to a severe accident of the nuclear installation, regardless of their low probability. This should be taken into account when reading this report.

Based on the facts identified during the accident at NPP Fukushima Daiichi, international nuclear institutions have issued a number of conclusions and lessons learned (for the nuclear industry and national nuclear regulators) applicable for all types of reactors. The submitted report summarized results of stress tests as specified in the declaration by ENSREG (European Nuclear Safety Regulators Group) of 13 March 2011 "EU Stress Tests Specifications". The stress tests are a part of a comprehensive evaluation of NPP safety, which is related to international documents published in connection with the given event, e.g.:

WANO	SOER 2011-2, Fukushima Daiichi Nuclear Station Fuel Damage Caused by Earthquake and Tsunami, March 2011.
WANO	SOER 2011-3, Fukushima Daiichi Nuclear Station Spent Fuel Pool/Pond Loss of Cooling and Makeup, August 2011.
INPO	Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Station, November 2011.
IAEA	International fact finding expert mission of the Fukushima Daichi NPP accident following the great east Japan earthquake and tsunami, 16 June 2011.
US NRC	Recommendation for enhancing reactor safety in the 21th century, 12 July 2011.

In its letter of 25 May 2011 SÚJB requested ČEZ, a. s. to perform the stress tests. The performance of the stress tests was regulated by an order issued by the director of the ČEZ, a. s. Production division which specified the scope and method of their execution.

The evaluation was performed by specialists in nuclear safety, designing of nuclear installations, accident management, emergency preparedness and research of phenomenology of severe accidents, who have been fully qualified for such activities. The evaluators proceeded in agreement with a deterministic approach with the anticipated gradual failure of all preventive measures during the evaluation of extreme scenarios.

Results of stress tests at Dukovany NPP and Temelín NPP, as a targeted evaluation of NPP safety margins and resistance requested by the European Commission, have confirmed effectiveness and correctness of resolutions adopted earlier about implementation of measures to increase resistance of the original design. No condition was found that would require an urgent solution. The plants are capable to safely handle even highly improbable extreme emergency conditions without endangering their surroundings.

3.2.3.2 Results of stress tests from the viewpoint of controlled severe accident management at NPP

Dependence on activities of other reactors on the site

Due to the independence of electric power supply of individual units from external and internal sources (including emergency sources) at both sites, the electric power supply sources of one unit may be conveniently used in case of SBO at the other unit.

Dukovany NPP

On the Dukovany NPP site there are 4 reactor units arranged into two double-units. Containments of the individual units of each double-unit are strictly separated during the operation and there are no risks of atmosphere leakage from one unit into the other. Fuel storage pools of the two units are situated in the shared reactor hall. In case of an accident during refueling it is necessary to address the issue of spreading of Ra substances in the shared reactor hall and into the open containment of the affected unit.

Reactors are completely independent in terms of technology, nevertheless many systems and auxiliary and supporting equipment can be shared, e.g. electric power supply, cooling water circulation, fire water, etc. The systems can be interconnected between all units.

ESW pumps from one main production building are situated in one CPS building, with electric power supply from the respective units but the delivery routes are shared by main production building so they can be used for both units of that main production building, i.e. 12 pumps supply ESW to 2 units.

Under emergency conditions the double-unit arrangement of the auxiliary systems enables replacement or making-up of media in the tanks of safety systems (ECCS) from the adjoining unit. If only one unit in a double-unit is affected it is possible to use water in the passive emergency system XL of the adjoining unit, which may represent at least 1000 m^3 of H_3BO_3 solution.

Temelín NPP

At the Temelín NPP site there are 2 units, independent in terms of technology and separated in terms of construction. Each Temelín NPP unit has SFSP in its respective containment.

The equipment shared by both units includes supplying with raw water from the Vltava River and CBSS for the transfer of heat from RC, SFSP and equipment of safety systems into the atmosphere as the ultimate heat sink.

Apart from CBSS (passive, seismic resistant objects), all other technological systems for heat transport are mutually independent and structurally separated for both units.

Other shared equipment, which may be significant for severe accident management, is the supply of boric acid solution kept in the auxiliary building of active operations for both units. It represents an additional supply of 1600 m^3 available for both NPP units (the volume is comparable with the quantity of boric acid solution available in the containment sump).

Potential effects of other equipment in the site proximity, including assumption of a limited capacity of personnel trained for accidents management at several units at a time

No such equipment is situated in the proximity of Dukovany NPP.

In the proximity of Temelín NPP units, in the minimum distance of 900 m, there are three branches of a transit gas line. It has been demonstrated that in case of simultaneous rupture of all three gas lines across the full diameter, with a subsequent gas outburst and ignition, the effect will not adversely affect the equipment that ensures safety of Temelín NPP units.

The capacity of the shift personnel at both sites is sufficient for the initial activities; however, for a long-term management of emergency conditions at all units at the same time the personnel shall be deployed in a special regime (alternating and additional personnel at highly exposed workplaces, time for rest, food and management of available resources).

Loss of communication equipment or systems

Backup power supply for operation of communication means used for warning of personnel on the site and for communication of the key personnel (ECC, shelters, LFRS, SÚJB, IRS, MCR personnel) is in case of a loss of power supply or damage of the infrastructure ensured mostly in hours. Sirens on the objects have no backup power supply. The site address system has no backup power supply. Sirens in the objects have their own accumulator batteries. The operating address system has a backup power supply.

In case of a long-term SBO a loss power supply to telephone switchboards of cooperating network workplaces outside Dukovany NPP or Temelín NPP may occur, except the central control center of the Czech Transmission Grid (CTG) Praha and the backup control center of CTG Ostrava that have their own DGs.

The recovery of power supply from sources outside Dukovany NPP (e.g. from hydropower plants Dalešice or Vranov) or Temelín NPP (from hydropower plant Lipno) is conditional on cooperation (communication links) between several external entities (ČEZ, CTG, E.ON).

In case of infrastructure damage the communication between the intervening individuals and control centers might by disrupted, as well as communication with external state administration bodies (crisis management teams of SÚJB and of the respective region, IRS, etc.), because availability and operating time of the existing communication means is fairly limited. Fixed telephone network, mobile telephone network, radio sets, warning means etc. are not secured against extensive damage of the infrastructure. However, the communication via radio sets used by LFRS to communicate with other IRS parts remains available all the time.

Worsening of performed works as a result to high local radiation intensity, radioactive contamination and destruction of some equipment on the site

In case of NPP damage the use of supporting and alternative technical means would be addressed through the established mechanisms of the emergency response organization. If the emergency control center cannot be used for any reason whatsoever then backup centers are available (for Dukovany NPP in Moravský Krumlov and for Temelín NPP in České Budějovice) with a limited quantity of information necessary for management of extraordinary events.

In case the plant is not accessible the situation would be addressed by limited alternation of the personnel and the personnel would sleep on the site or its close proximity (in shelters and ECC, possibility to use the building of the information center).

Each shelter at NPP is provided with equipment for protection of persons against effects of radioactive substances, warfare poisonous gases and biological substances. In terms of the construction, the shelters are designed to provide protection to persons against effects of penetrating radiation. Technical equipment of the shelter enables their operation for at least 72 hours (including food, drinks and hygiene). The basic equipment of the shelters includes dosimetric devices to measure surface contamination and dose rate, a supply of spare emergency protective equipment, spare clothes, iodine prophylaxis means, means for communication with the emergency response board (ERB) workplace. The distribution of spare emergency protective equipment and sanitary material is performed by members of the shelter team based on justified needs and requirements of sheltered persons.

Although no heavy technology is available directly on the site to remove debris from the backbone and access roads that might be blocked with debris from non-seismic resistant objects, intervention instructions of the emergency response organization (OER) specify potential use of the means through IRS.

Effect on the accessibility and habitability of the main control room and emergency control room and measures to eliminate or to manage such a situation

The equipment of the main control room (MCR) and emergency control room (ECR) is situated in the room adjoining on the containment. An order to use breathing devices in MCR is in the competence of the Unit Shift Supervisor (USS). Provided MCR cannot be used then the operative management personnel will be moved into ECR in a controlled manner. The relocation of the personnel from MCR to ECR may be in justified cases also decided by SE or USS (or Safety Supervisor). ECR may be used to monitor operating parameters and to control components of safety systems in a similar scope as in MCR.

MCR rooms and, to a lesser extent also ECR rooms, could be affected by radiation at a high pressure and, at the same time, at high radiation doses inside the containment or at high releases of fission products from the containment.

MCR and ECR at Temelín NPP are equipped with filtrating ventilation systems power supplied from UPS safety systems, so they are habitable even in cases of anticipated leakage of fission products. Dukovany NPP has a project that has been approved and prepared for implementation to equip MCR and ECR with additional filtration and ventilation systems. At present, a threat to the personnel would be addressed by evacuation of MCR personnel, based on an order of ERB commander, based on evaluation of the radiation situation when a criterion specified in an action instruction is fulfilled (subsequently, only short term entries would be considered to perform actions).

Feasibility and effectiveness of measures for management of accidents under external risk conditions

No acute shortcomings have been identified in respect to fulfillment of safety functions as a result of external threat to NPP:

- Procedures and strategies (EOPs) have been developed in order to ensure fulfillment of safety functions for the stage before RC damage and for the stage after fuel damage in RC (SAMG). Due to the symptom-based approach to dealing with emergency conditions their applicability is not limited to consequences of external conditions.
- Neither EOPs nor SAMG (with the exception of LFRS means at Dukovany NPP) anticipates any other mobile or non-technological means or deliveries. A potential use of supporting and alternative technical means would be operatively addressed by using mechanisms of the emergency response organization. Documentation for management of emergency condition in the emergency control board (ECB) and technical support center (TSC) uses the assumption of access to data in ECC or TSC. However, no documents have been developed for cases of activation of emergency control board and TSC in other locations.
- NPP personnel is sufficiently qualified and trained to use EOPs and SAMG, as well as to perform evaluation of damage of the equipment after seismic events. The personnel

are also trained to perform manipulations to bring supply from internal or external sources in case of SBO.

- In respect to IOER or SOER shift personnel no shortcomings have been identified concerning the number of personnel necessary to mitigate the above-mentioned consequences of beyond-design basis events.

In case of an extensive damage of infrastructure and long-term unavailability of the site (collapse of buildings, damaged communications etc.) the alternating personnel would get to the site only with difficulties. In that case the required activities would have to be provided for by the personnel present on the site at the time of the event. However, the network of access roads and bridges over waterways in valleys in the plant´s proximity is so dense that it is practically certain that the access to the plant will be always possible from one direction. The alternation of personnel would be dealt with operatively, in cooperation with state administration bodies (IRS, Czech army, etc.).

It would not be probably possible to use shelters of emergency preparedness or the emergency response board workplace or technical support center, which are situated under seismically non-resistant objects. Activities of TSC and ERB would be in this case performed operatively.

Unavailability of energy supply

At both sites, Dukovany NPP and Temelín NPP, the existing capacity of accumulator batteries SPS category I might endanger performance of some interventions and disable some measurements. The time may be extended by a controlled disconnection of non-essential appliances. These activities have been already included into the existing EOP and SAMG. The scope and the order of the equipment and components which would be disconnected to reduce the battery load depends in their importance in respect the ongoing emergency event and the employed strategy. The purpose of relieving load of the accumulator batteries is to ensure the longest possible functioning of I&C and PAMS systems (control and monitoring of parameters) and power supply of equipment necessary for execution of essential safety activities (start of DG and renewal of power supply, isolation of routes that remove coolant from I.C., regulation of pressure in SG and in I.C., isolation of the containment, etc.).

SAMG anticipates the option to use portable electric LFRS control boards to control some drives directly from the switchboards.

Management of hydrogen risks in the containment

Containments of Dukovany NPP and Temelín NPP units are equipped with a system for liquidation of post-accident hydrogen which is proposed only for design basis accidents. This system contains passive autocatalytic recombiners and it is able to liquidate in the long-term the hydrogen released during design accidents and post-accident conditions and thus to maintain concentration of hydrogen at values at which it may ignite only under conditions of design basis accidents. The existing system of hydrogen liquidation may not be sufficient for severe accidents. However, design preparations have been currently under way for installation of a system for liquidation of hydrogen generated in severe accidents.

Potential failure of measuring and information systems

Most of the required information about the condition of components and values of parameters necessary for severe accident management are available in PAMS.

All systems important from the viewpoint of safety are qualified for design basis accidents and post-accident conditions. They are not qualified for conditions of severe accidents but in many cases the measuring scale takes into account requirements for management of an early stage of severe accidents.

A limited set of parameters is used to diagnose the accident condition and to verify implementation of the selected strategies. Measured values of selected quantities from standard instrumentation are used for verification. Several quantities are determined for each parameter which may be used for its verification (value, trend). Always a direct measurement of the required parameter is used and one or several measurements of alternative quantities, which may be used to derive the value or trend of the required parameter. In some cases during severe accidents it is impossible to evaluate the value or the trend of the required parameter based on directly measured values, either due to their unavailability or absence of the measurement of the given parameter. In those cases computation tools are used to determine the required parameter (simple diagrams of parameter dependence). The inputs into the computation tools may be either directly measured values or previously specified, defined values.

The ability of the measurement systems to survive in conditions after a severe accident has not been verified but they are expected to be sufficiently robust to resist conditions of a severe accident at least for a certain time.

3.2.3.3 Results of stress tests of controlled severe accident management from the viewpoint of Dukovany NPP technology

Activities after fuel damage in the reactor pressure vessel

The fundamental cause of severe accidents is insufficient removal of residual heat released from fuel in RC. A beyond-design damage of RC means a local exceeding of the temperature of cladding 1200 °C, when the steam-zircon reaction develops. As this parameter cannot be measured a set point was specified the transition to SAMG as the temperature of 550 °C at the output from RC. Exceeding of 1200 °C in an extensive area leads to an intensive steam-zircon reaction, which is exothermic. This means that a much greater quantity of heat than residual heat will be quickly released and this heat will contribute to the accident development because it mostly accumulates inside RC.

The renewal of heat removal from RC on II.C. side with alternative means is performed in agreement with EOP, i.e. before transition to SAMG. Further, activities associated with depressurization of I.C. are performed with the objective to enable injection by low-pressure pumps into I.C.

There are two permanent methods to prevent a loss of cooling of RC from developing into a severe accident:

- recovery of heat removal via SG (alternative making-up of SG from low-pressure sources, including making-up of water with LFRS means)
- heat removal by making-up of coolant into I.C. and its draining through an escape opening in the primary system (during LOCA) or through open valves in the pressurizer (feed & bleed).

EOPs also include alternative strategies:

- depressurization of the primary system or cooling from II.C. side, which may lead to initiation of the hydro accumulators or even low-pressure emergency or alternative sources,
- recovery of availability of high-pressure systems of emergency making-up or alternative high-pressure systems of emergency making-up into I.C.,

- use of the remaining coolant in the loops with a forced start of MCP, even if it may lead to its destruction.

When using a conservative approach, the fuel damage may be linked with the start of the steam-zircon reaction associated with a massive production of hydrogen, which precedes the start of a loss of RC geometry. Symptoms of RC damage by melting include, apart from the continuously growing temperatures, particularly growing concentration of hydrogen in the containment. With regard to the rate of hydrogen generation before the loss of geometry, the concentration of hydrogen may not be manageable sufficiently quickly by the existing recombiners. However, there is still a sufficient time for potential safe ignition hydrogen in the early stage (tens of minutes).

A typical time from the entry into SAMG to damage of RPV integrity by the melted core is ca. 7 hours on the condition that all methods of coolant delivery into the vessel have failed.

The strategy for renewal of heat removal is addressed in SAMG by means of depressurization and particularly making-up of I.C. In this stage of the accident it is no more possible to use cooling of I.C. from the II.C. side and therefore the coolant needs to be delivered directly into the reactor vessel. SAMG recommends to start delivery of water at the moment when the source is recovered to supply a quantity greater than the minimum flow rate necessary for RC flooding. The minimum flow rate specified in SAMG is the flow rate evaporated by the residual heat in RC.

The risk of vessel failure would be significantly reduced by implementation of the strategy to cool the vessel from outside by flooding the reactor cavity. The completed analyses have shown that the VVER 440/213 design is convenient from the viewpoint of keeping the melt inside the reactor vessel and its cooling from outside, although the original design did not anticipate this measure. Particularly the residual power of the reactor is very low which guarantees low heat flows on the outer surface of the vessel in the area of bubble boiling with a big boiling margin. The vessel has no penetration in its bottom part. The reactor cavity ranks among the lowest places in the containment and in case of a loss of water for emergency cooling it is sufficient to discharge bubbler condenser trays to flood it.

The decision about the increase of resistance of Dukovany NPP design in respect to severe accident management was adopted after completion of the Periodic Safety Review in 2006. Some adjustments have been already completed on Dukovany NPP units that lead to cooling of the vessel from outside. The adjustments include primarily closing of the drain at the cavity bottom, a measurement of water level added in the reactor cavity room and adapted intakes of the ventilation route TL11 into the reactor cavity room, including preparation of suction openings so that they can be equipped with inlet valves. The measures to be completed include certain adjustment of insulation in the bottom part of the vessel, so that it does not prevent water from getting to the vessel, and minor adjustments in the bottom part of the cavity room (screens) and in the upper part of the room (steam removal into the containment from the reactor cavity). All those activity support the strategy mentioned above.

Activities performed after disruption of the reactor pressure vessel

The performed analyses have shown that if the accident inside the reactor pressure vessel cannot be stopped the vessel would fail in its bottom part. After RPV damage the material would relocate from the vessel and gradually it may form a dense layer of debris which might be hard to cool. The debris would melt through and interact with concrete even under a potential water layer which would be unable to remove heat because it might be isolated from the debris by film boiling. The main consequences of this stage of the accident could be as follows:

1. additional production of hydrogen from non-oxidized Zr, steel in the debris and concrete reinforcement,
2. penetration of the melt through the reactor cavity wall,
3. production of hydrogen from interaction of the melt with the concrete of the reactor cavity bottom is much slower than in case of cladding oxidation. It is by two orders of magnitude lower than the production of hydrogen from the reaction of water steam with zircon cladding.
4. The penetration of the melt through the cavity wall is more serious than penetration through the cavity bottom because:
   - the penetration of the melt through the wall in the radial (horizontal) direction is faster than penetration in the axial (vertical) direction,
   - the wall (2.5 m) is thinner than the bottom (3.1 m),
   - the cavity wall forms the boundary of the containment; the debris penetrates the bottom and gets into the base plate (sub-base), where the fission products are retained.

In a VVER 440 reactor the door in the cavity would be protected against contact with liquid debris for some time with tough debris or crust. In any case it is not possible to exclude a small damage of the containment shortly after the vessel bottom failure as a result of a failure of rubber gasket on the door.

The door could be protected by flooding of debris in the cavity. Even with damaged gasket there is still the gasket of the outer door which might prevent leakage of water and thus protect the door. This method of door protection has not been analyzed and all these considerations are based on an expert estimate.

Provided all measures are adopted to prevent the door failure then the melt penetration through the cavity wall may occurs only ca. 4 days after the vessel bottom failure. This represents a major late damage of the containment. At that time the concentration of fission products in the containment would be already low.

The strategy of cooling of the melt is a part of SAMG “Flooding of the cavity“. The current configuration of the plant provides the opportunity to flood the cavity by spillway but this requires water from two TH tanks (low-pressure ECCS) and bubbler condenser trays. The SAMG therefore considers discharging of bubbler trays, including a check whether the drainage of the box is closed in order to prevent more losses. The strategy also anticipates the use of water from the adjoining unit.

The main contribution of the strategy of flooding the debris in the reactor cavity is the cooling of the steel door and interception of fission products released from the interaction of the melt with the concrete.

Management of risks of hydrogen presence in the containment

Containments of Dukovany NPP units are equipped with a system for liquidation of post-accident hydrogen designed only for design basis accidents. For design basis LOCA accidents that produce only a very small quantity of hydrogen there are 17 recombiners available situated inside the containment.

The containment integrity in an early stage of a severe accident is threatened most by a big fire or hydrogen detonation and by a subsequent failure of the double door in the reactor cavity. In a late stage of the accident the situation is aggravated by penetration of debris through the cavity. The containment may be endangered by hydrogen after the beginning of

RC damage during the steam-zircon reaction. Due to the large surface of the cladding and exothermic character of the reaction the hydrogen is generated very quickly, from 0.5 to 1 kg/s. With regard to the rate of hydrogen generation before the loss of geometry the quantity of hydrogen cannot be managed by the existing recombiners. Also the RC debris potentially expelled from the reactor vessel is a major source of hydrogen. An intense hydrogen production occurs ca. 30 minutes after the temperatures of the gas at the outlet from RC exceeds 550 °C. The course of hydrogen development from the steam-zircon reaction is substantially more intense at a high pressure and therefore one of the first requirements in SAMG is the instruction to depressurize I.C.

A resolution to increase resistance of the Dukovany NPP design in respect to severe accident management was adopted after the Periodic Safety Review in 2006. A project to develop a system for effective liquidation of post-accident hydrogen is in the final stage of preparations, which will be capable of handling even the hydrogen hypothetically generated in the worst case scenario (from the viewpoint of hydrogen generation) of a severe accident. The analyses completed by now and experience from other VVER have confirmed that the system made of high-performance recombiners (ca. 30 pcs), complemented with igniters in case of spray functioning, is able to reduce the risk of flame acceleration and exclude the risk of transition to detonation.

Cliff-edge effects in the delay between reactor shutdown and core melting

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 cavity.

The longer lasting loss of cooling of RC could undermine the integrity of RPV by melted core. This moment is characterized by the end of in-vessel phase of severe accidents and the beginning of ex-vessel phase. After the melt gets from the RPV to the bottom of the containment an interaction starts between the melt and the concrete. The slower phenomenon is the melting of the foundation slab. It is thicker than the wall of the cavity, 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 side penetration of debris through the wall of the cavity, 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 cavity after the penetration of debris through the vessel could extend this time and it could protect the steel door.

Despite measures in SAMG for accident management with the objective to prevent a loss of containment integrity, opportunities have been identified to improve the ability to maintain containment integrity after a serious damage of fuel, which consist in a proposal and implementation of other means to ensure containment integrity (i.e. prevention of release of fission products) during a severe accident. The means may include particularly the system for liquidation of hydrogen in the containment and measures for localization of the melt on the containment bottom.

The opportunities for improvement of defence in-depth during events that may lead to a severe accident are provided in the table below. The table contains also the areas where additional analyses are required because they were not available at the time when the evaluation was performed.

Some of the measures (in the note marked as “ PSR finding“) would have been implemented even without this targeted evaluation, whose outputs only confirmed effectiveness and correctness of the earlier adopted resolution to implement measures that improve resistance of the original design.

Tab. 2: The opportunities for improvement of defence in-depth during events that may lead to a severe accident 1

Opportunity for improvement	Corrective measures	Deadline (Short-term I / Medium-term II)	Note
Containment integrity during a severe accident	Increase of the capacity of the system for liquidation of post-accident hydrogen	II	PSR finding
Localization of the melt in the reactor core	Cooling of the melt from outside of RPV	II	PSR finding

Management of accidents after uncovering of the upper part of the fuel in the storage pool

The Dukovany NPP fuel storage pools are situated in the reactor hall shared by two units. An analysis of the course of accidents in the storage pools for shutdown conditions has been planned for 2012. It will analyze behavior of the pool in the mode 6, i.e. during refueling, in the mode 7, i.e. during total removal of fuel from the reactor, and in the modes 1 through 5, when the storage pool and the reactor hall are hermetically separated from the containment.

Because the storage pools are not situated in hermetically separable premises (only in the reactor building shell) this would be followed by a release of radioactive substances into the plant´s proximity. In case of the steam-zircon reaction hydrogen would be released into the reactor hall.

The risk of hydrogen in the reactor hall, i.e. outside the containment, has been evaluated and results of the analyses have shown that even if cooling in both pools fail this would not probably lead to a concentration of hydrogen in the reactor hall that might reach the combustion limit for hydrogen.

Limitation of leakage after a severe damage of spent fuel in the storage pools

No procedures to deal with accidents associated with a meltdown of fuel in SFSP are available yet. Although the personnel in MCR or TSC do not have the so-called shutdown SAMG (SAMG for shutdown conditions), available options are known and they consist in continuing making-up of water and heat removal and potential isolation of leaks from SFSP in agreement with EOPs. The damage would occur after a relatively long time, with the exception of mode 7, which provides a sufficient time to find an operative solution.

The principle measure to limit release into the surrounding environment is to stop or to slow down the accident by flooding SFSP with water. An emergency system is being prepared for pool flooding which will be combined with other measures in the reactor hall that exclude presence of any operating personnel.

The reactor hall has big volume which has a positive effect on dilution of the fission products. Other potential measures limiting the release are as follows:

- In case of activity release from SFSP (or from the reactor in the mode 6) it is necessary to immediately switch off large-capacity systems for ventilation of the reactor hall; the procedure is already provided in the existing EOP for shutdown conditions.
- Once all the personnel leave the reactor hall it is essential to tightly close all access routes for the personnel into the reactor hall.
- If the unit is in the refueling mode or in case of total removal of fuel from the reactor, when the containment is usually connected with the reactor hall via several passages, it is necessary to switch off ventilation systems of the containment, to make sure that all persons leave the containment and to quickly close all access routes into the containment of the unit in the modes 6 or 7. These measures are necessitated by the fact that it is impossible to quickly separate the containment from the reactor hall.

Measurements that characterize the condition of SFSP (temperature, level, flow rate in cooling system) are available on panels in MCR. Measurements of parameters associated with cooling of SFSP are not available in ECR or in PAMS. Similarly, PAMS does not have measurements about the Ra situation in the hall in the proximity of SFSP. Due to the big volume of the reactor hall, its lower tightness and low residual power of the fuel, the anticipated conditions are not as severe as inside the containment. Most measurements will therefore remain available. The most important of them are measurements of activity in the atmosphere and water level in SFSP.

Although the main objective of SAMG is to prevent the loss of containment integrity, as the last barrier against release of fission products into the surrounding environment, along with limitation of release of fission products, SAMG also describes strategies that use all available means to stop or at least reduce of release of fission products after the loss of containment integrity.

Respective instructions will be developed for a systematic use of all available possibilities to limit leakage from the reactor hall.

3.2.3.4 Results of stress tests of controlled severe accident management from the viewpoint of Temelín NPP technology

Activities after the fuel damage in the reactor pressure vessel

Measures for management of accidents after a severe damage of fuel are described in SAMG strategies that use all available means for making-up I.C. in order to recover reactor core cooling. Each individual system for making-up of I.C. is able to supply a sufficient quantity coolant for removal of residual heat from the damaged fuel, although the flooding of RPV from inside does not guarantee reactor core cooling because the melting core may get into a condition when the cooling is no more possible.

All strategies are based on the principle of cooling of the damaged fuel inside of RPV, i.e. making-up of water into I.C. With regard to the thermal output of the reactor and design solution of the concrete reactor cavity, no possibility of RPV cooling from outside has been currently known for VVER 1000 units with V320 reactors. This fact will be a subject of further analyses.

Reactor core cooling in the phase after a severe damage of fuel is renewed by means of activities described in SAMG. The following strategies have been defined to renew RC cooling:

- Making-up of water into a hot dry reactor core will always positively affect consequences of the accident. An optimum method to resume making-up of I.C. has been specified to minimize subsequent release of fission products into the atmosphere.

- Another measure after a severe damage of fuel is depressurization of I.C. The purpose of the depressurization is to reduce the pressure in I.C. below the value under which no direct heating of the containment may occur because the melt is not ejected from the reactor under high pressure. There are several methods of I.C. depressurization (use of the system of emergency I.C. venting, pressurizer relief valve, normal injection into the pressurizer, SG depressurization, etc.).

Activities after disruption of the reactor pressure vessel

After RPV failure the reactor core debris moves into the concrete reactor cavity or other parts of the containment. If there is no water in the containment the reactor core debris will start attacking the containment concrete bottom and an interaction between the melt and the concrete will occur which will result in generation of hydrogen and other non-condensable gases. The measures for management of accidents after a severe damage of fuel and relocation of the melt to the containment bottom are described in SAMG strategies that use all available means to make up the containment in order to ensure cooling of the melt.

All strategies to cool the melt on the containment bottom are based on the principle of flooding melt from above. Flooding of the reactor core debris outside of the reactor pressure vessel with water ensures heat removal from the debris and reduces the rate of concrete attack.

One of the outputs of analyses of sequences leading to severe accidents, as selected based on PSA Level 2 results, was the time before disruption of RPV integrity by RC melting. For the least favorable scenario, on condition that all methods of coolant delivery into RPV fail, the time may be ca. 4.5 hours. This moment characterizes the end of an in-vessel phase of a severe accident and the beginning of an ex-vessel stage, with all accompanying phenomena in the containment (interaction of the melt with the concrete resulting in hydrogen generation, direct heating of the containment, etc.).

Water is made up into the containment as a preventive measure in case of a severe accident. The respective strategy in SAMG provides instruction for containment flooding with water up to the maximum measurable level which will ensure both protection of the concrete on the containment bottom in case that RC debris gets from RPV into the containment and effective washing–out of fission products released from the melt.

If the molten core-concrete mixture is flooded with water, the heat transfer from the upper surface of this mixture will be significantly more efficient due to water boiling.

In the course of a severe accident the containment can be made up with standard making-up methods from supply tanks for refueling or from primary coolant hold-up tanks with and also using an alternative method of making-up with stable fire pumps or overfilling of bubbler tanks.

Management of risks of hydrogen inside the containment

The design function of the containment is to limit potential radiation consequences of a potential accident on the reactor equipment and this means that the containment is the last barrier against release of radionuclides from fuel or coolant in I.C. in case of an accident.

The existing measures for management of accidents with a threat posed by hydrogen to containment integrity are described SAMG strategies that use all available means that prevent dangerous forms of hydrogen burning. The two regimes that are the most dangerous for containment integrity are hydrogen burning – fast deflagration and transition from fast deflagration to detonation. In order to assess the risks, analysis were made of time curves of spreading and distribution of hydrogen generated during severe accidents in the entire containment. The Temelín NPP unit containments are equipped with a system for liquidation of post-accident hydrogen intended for design basis accidents.

Still, the existing system of hydrogen liquidation may not be sufficient for a severe accident. At present, however, design preparations have been under way for installation of a system for liquidation of hydrogen generated during severe accidents.

Analyses have been conducted with the objective to determine the limit case of pressure increase in the containment.

The analyses have shown that after a melt through of the reactor pressure vessel (RPV) and relocation of the melt to the bottom of the concrete reactor cavity the pressure in the containment cannot increase to values that might seriously threaten its integrity. Further increase of pressure in the containment above the values that threaten its integrity may occur only after the beginning of the interaction between the melt and the concrete in the ex-vessel phase.

The value of the pressure that may disrupt containment integrity is approximately 1.6 multiple of the design pressure.

The performed analyses have shown that cooling of the pool of melted material in the cavity with water may reduce the rate of concrete decomposition and thus to postpone potential failure of the containment until a late phase of the accident. Decelerated decomposition of the containment base plate will postpone or completely stop the massive release of radioactive substances into the external environment after the containment bottom is melted through. It will also exclude a large scale steam explosion and the reduced thickness of the melt layer increases probability that the melt will be cooled down and that decomposition of the concrete with the melted material will be stopped.

A modification was implemented consisting in closing of channels with ex-core neutron flux measurements passing through the containment bottom in order to improve resistance of the containment bottom in the concrete reactor cavity against melt through with a release of the melt after RPV failure. The channels were closed with removable steel casings filled with refractory material. This solution provides for a high resistance against the melt penetration and at the same this it does not influence instrumentation for neutron flux measurement.

Cliff-edge effects in the period between the reactor shutdown and core meltdown

Analyses of SBO scenarios with a loss of heat removal from I.C on the SG side indicate that without alternative activities described in EOPs there is only a very short time reserve for recovery of heat removal from I.C. The temperature of 650 °C at the output from the reactor core could be in the least favorable case achieved in ca. 2.5 to 3.5 hours after SBO. If the temperature at the output from RC exceeds 650 °C and continually increases, it has a character of “cliff edge“ condition from the viewpoint of damaged fuel in the reactor core.

In case of a long-term loss of RC cooling the integrity of RPV might be damaged by the melted core. For the worst-case scenario, provided all methods of coolant delivery into the RPV have failed, the time may be ca. 4.5 hours. This moment characterizes the end of an in-vessel phase of a severe accident and the beginning of an ex-vessel phase.

After the melt gets from RPV to the containment bottom an interaction starts between the melt and the concrete. The result of the interaction is a decomposition of concrete with an intense production of hydrogen. Consequently, the containment bottom will become thinner and once it reaches a value at which the remaining concrete would break under the weight of the melt, the melt penetrates the lower, non-hermetic part of the reactor building. In the least favorable case, when all possibilities of cooling the melt fail, the penetration of the melt into the lower non-hermetic part of the reactor building reactor might occur ca. 24 hours after the accident.

Despite measures in SAMG for accident management with the objective to prevent a loss of containment integrity, opportunities have been identified to improve the ability to maintain containment integrity after a serious damage of fuel, which consist in a proposal and implementation of other means to ensure containment integrity (i.e. prevention of release of fission products) during a severe accident. The means may include particularly the system for liquidation of hydrogen in the containment and measures for localization of the melt on the containment bottom.

The opportunities for improvement of defence in-depth during events that may lead to a severe accident are provided in the table below. The table contains also the areas where additional analyses are required because they were not available at the time when the evaluation was performed.

Some of the measures (in the note marked as “PSR finding“) would have been implemented even without this targeted evaluation, whose outputs only confirmed effectiveness and correctness of the earlier adopted resolution to implement measures that improve resistance of the original design.

Tab. 3: The opportunities for improvement of defence in-depth during events that may lead to a severe accident 2

Opportunity for improvement	Corrective measure	Deadline (Short-term I / Medium-term II)	Note
Technical means	System of liquidation of hydrogen in the containment for severe accidents	II	PSR finding Preparation of change of equipment configuration
Analysis	Localization of the melt outside RPV	II	PSR finding To be addressed in coordination with other VVER 1000 operators

Management of accidents following uncovering of the upper part of fuel in spent fuel storage pools

In Temelín NPP units SFSP are situated in containments. Provided SFSP contain spent fuel it is necessary to maintain sufficient supply of the coolant and to ensure removal of the released heat.

An alternative method of heat removal from SFSP in case of a loss of normal cooling is the emergency cooling SFSP by means of the containment spray system.

Computations have been conducted to analyze the loss of SFSP cooling with the stored spent fuel. Results of the computations are the maximum temperatures achieved in SFSP during storage with cooling, heat-up trends and margin to saturation temperature and time before the heads of the stored fuel assemblies become uncovered , following the loss of SFSP cooling.

To ensure shielding of radiation from the fuel assemblies the level shall not drop below 783 cm. The location of SFSP in the containment ensures that no undesired irradiation of persons occurs even after the level in SFSP drops below the value necessary for shielding of radiation from the spent fuel.

Because in the modes 5 and 6 (cold condition and outage) the hermetic closures of the containment may be open and workers may be inside the containment to perform some works during the outage, the health of such persons may be at risk. Therefore one of the requirements to be met immediately after identification of an emergency situation in the respective procedure is the evacuation of all workers present in the containment during the event and closing of hermetic closures.

Limitation of leakage after a severe damage of spent fuel in storage pools

Technical means for mitigation of consequences of damaged fuel in SFSP are available and the strategies consist in a continuing making-up of water into the containment and heat removal and potential isolation of leaks from SFSP in agreement with EOPs. Shutdown SAMG for accidents associated with melting of fuel in SFSP have not been developed yet.

No analyses have been conducted for damaged spent fuel in SFSP. With regard to the existence of an alternative method of making-up SFSP with the containment spray system no long-term loss of heat removal from SFSP is expected without a simultaneous loss of heat removal from the reactor core.

The key parameter for evaluation of a loss of heat removal from spent fuel stored in SFSP is the water level in SFSP. As long as the fuel is covered with a layer of water (even when boiling) the residual heat will be removed from the fuel. Once the water boils out from SFSP and once the stored fuel assemblies are exposed, they will start overheating. The level in SFSP and several other parameters, such as the status of ESW systems and flow rate of ESW into the heat exchanger for SFSP cooling, are communicated via PAMS. Another parameter measured in SFSP is temperature.

3.3 DESCRIPTION OF ACTIVITIES PERFORMED BY THE STATE REGULATORY AUTHORITY

3.3.1 Overview of implemented and planned activities

Both plants have implemented practically identical systems for severe accident management to ensure the 4th level of the defence in-depth and the system of emergency preparedness to ensure the 5th level of defence in-depth. Functioning and interconnected system for management of accidents and emergency preparedness are at both power plants ensured with a set of robust measures of personnel management, administrative and technical nature. The system is systematically monitored and inspected by the state regulatory authority.

In the personnel area the regulatory authority supervises organization of emergency response and assurance of activities to be performed by individual functions; in the administrative area the authority supervises implementation of the respective procedures, guides and instructions and the use of capacities of technical supporting centers; in the technical area it supervises assurance of function of the requested scope of technical means for implementation of the strategies. The authority also monitors practicing of interventions in case of extraordinary events in the first (preventive) phase of event development by the personnel in continual shift operation. The practice also includes situations when the event scope grows beyond the capacity of the personnel in continual shift operation and when the second phase starts (mitigation of consequences), during which the emergency response organization is activated. Subsequently, the authority monitors takeover of responsibility for management of interventions by the plant´s emergency response board supported by the Technical support center.

It is required that all necessary activities during extraordinary events shall be managed and performed from protected places. TSC and ERB that manage strategies under SAMG are situated in the emergency control center (ECC), which has been designed as a secured workplace habitable also in case of activity release into the atmosphere. In addition to the regulatory monitoring mentioned above the following activities are subject of regulatory body monitoring, as well:

- activities for implementation of strategies performed by the shift personnel from MCR or ECR
- practicing of procedures to deal with technological accidents based on strategies contained in EOPs, while the main priority is to renew heat removal from the reactor core and to prevent damage of the 1st barrier against release of fission products (fuel cladding)
- transition of strategies for mitigation of consequences of severe accidents contained in SAMG, while the main priority is to prevent damage of the 3rd barrier against release of fission products (containment), which is the last intact barrier at that time
- updating of EOPs and SAMG to include findings from practicing of their use on the simulator or during emergency drills and external findings.

SÚJB is involved in practicing of situations associated with a response to an extraordinary event. When declaring a particular level of an extraordinary event (Alert, Site emergency, General emergency) the authority evaluates activation of emergency response organization, with an internal part (IOER) consisting of shift personnel, and standby emergency response organization (SOER) consisting of NPP specialist technical personnel who are on standby.

A system of qualification requirements is in place to select the shift personnel and workers for SOER and additional criteria are considered that take into account their knowledge and expertise. Preparedness of the shift and technical personnel to manage technological accidents is regularly verified during training on the full-scale simulator in presence of TSC personnel and during the emergency drills.

The organizational method to manage extraordinary events (including severe accidents) is at both power plants specified in the internal emergency plans approved by SÚJB.

Local fire rescue services (LFRS) are available at both sites in agreement with the legislative and regulatory requirements, with appropriate firefighting techniques and training to intervene at any part of the site. The pumping equipment of the LFRU belongs to the major mobile non-technological means available for transport and pumping of fluids.The program of accident management at Dukovany NPP and Temelín NPP is provided with an analytical support. The analytical support is based on probabilistic - deterministic approach which consists in selecting the most probable emergency scenarios leading to severe accidents and subsequent deterministic analysis using integral computation codes. The result of the analytical support is a summary of findings that consist in understanding phenomena during severe accidents and their timing, identification of potential weaknesses of the design, determination of activities for mitigation of consequences of severe accidents, validation of activities to respond to severe accidents and identification of a source term for evaluation of potential radiological consequences. Also a simulation tool is available to display phenomena for specific scenarios of severe accidents.

VVER 440/213 reactors in operation at Dukovany NPP have a specific design of the containment equipped with a passive condensation system (bubbler condenser system) and its basic function is to reduce pressure in the mixture of air and water steam in the reactor hermetic zone following the maximum design basis accident (guillotine break of the primary piping with the diameter 500 mm) by condensation of water steam in special trays filled with H_3BO_3 solution, with a subsequent isolation of non-condensed gases in hermetic air traps with check valves. At the same time, by creating under pressure against the surrounding atmosphere, the system minimizes potential releases of radioactivity outside the hermetic premises.

Due to limited information about the experimental verification of the system from authors of the original design and due to the need to expand knowledge and ability to model integral behavior of the system and partial physical phenomena in the conditions of large and small LOCA accidents, numerous international projects and studies were organized jointly in 1990s by countries operating those types of reactors. The projects were supported by national regulatory authorities of the countries that use units of VVER 440 type and in some cases they were also coordinated by the national regulatory authorities.

The completed studies provided a positive, factual and objective evidence that the bubbler condenser system is qualified for the conditions of the maximum design basis accident at VVER 440/213 reactors and that it performs its safety function under those conditions – keeping the integrity of the reactor hermetic zone and prevention of release of radioactive substances into the environment. At the same time, the studies provided data for validation of computation codes used for modeling of processes in the containment during beyond-design basis and severe accidents.

Apart from the above-mentioned measures proposed by both power plants, SÚJB is considering a proposal to Temelín NPP to analyze the possibilities and variants of modification and completion of the design with the objective to introduce ventilated containment for severe accidents (type II). This solution has been already implemented in various forms at many NPPs in the West. The procedure should be coordinated with other operators and regulatory authorities of countries operating VVER – 1000 reactors.

SÚJB will further propose to ČEZ, a. s. to consider establishment of a joint center for operators of VVER reactors so that they can help each other in case of severe accidents (Dukovany, Jaslovské Bohunice, Mochovce, Paks) based on their mutual similarities. The joint center for severe accidents would enable to effectively solve the issue of expensive mobile Diesel generators, heavy technology and other equipment which with the utmost probability will never be used.

3.3.2 Further steps to be taken by the state regulatory authority

An analysis of PSR findings from evaluation of Czech NPPs following the events at NPP Fukushima Daiichi has shown that there are no new acute safety findings and that it is necessary only to consistently continue fulfillment of the prepared schedules and to complete or to finalize and to deepen the prepared measures.

SÚJB will make sure that the Licensee implements the respective measures specified in the approved schedules.

3.3.3 Conclusions by the state regulatory authority

As it has been described above, no problems have been found requiring an acute solution. The proposed measures have been elaborated to a certain level and there is a plan for further course of action for their performance and a schedule for their completion. The authority monitors activities of the Licensee in agreement with the plan of implementation of corrective measures from PSR or from additional evaluations within the stress tests.

3.4 FINAL SUMMARY OF CHAPTER 3

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 3.2.1) **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 3.2.2)** **Schedule Or Milestones for Planned Activities**	**(Item 3.2.3)** **Results Available** **- Yes?** **- No?**	**(Item 3.3.1)** **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 3.3.2)** **Schedule Or Milestones for Planned Activities**	**(Item 3.3.3)** **Conclusion Available** **- Yes?** **- No?**
Topic 3 **Management of Severe Accidents and Recovery of Safety Functions of Units on the Site**						
Increase capacity of the system for liquidation of post-accident hydrogen at Dukovany NPP	Ongoing	Medium-term 2015	No	Ongoing Regulatory activity	Medium-term 2015	No
System for hydrogen liquidation in the Temelín NPP containment for severe accidents	Ongoing	Medium-term 2018	No	Ongoing Regulatory activity	Short-term 2013	No
Dukovany NPP – Cooling of the melt from outside of RPV	Planned	Medium-term 2015	No	Ongoing Regulatory activity	Medium-term 2015	No
Temelín NPP – localization of the melt outside RPV	Planned	Medium-term 2018	No	Ongoing Regulatory activity	Medium-term 2018	No
Temelín NPP – increase of coolant supply in the containment be used for emergency making-up	Planned	Medium-term 2018	No	Ongoing Regulatory activity	Medium-term 2018	No
Dukovany NPP – adding measurements of RA situation and SFSP into PAMS	Ongoing	Medium-term 2015	No	Ongoing Regulatory activity	Medium-term 2015	No
Develop “Shutdown SAMGs“	Planned	Short-term 2013 NPP Dukovany 2014 NPP Temelín	No	Planned Regulatory activity	Medium-term 2013 NPP Dukovany 2014 NPP Temelín	No
Dukovany NPP – verification of the analysis “Protection	Ongoing	Short-term 2013	Yes	Ongoing Regulatory activity	Short-term 2013	Yes

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 3.2.1) **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 3.2.2)** **Schedule Or Milestones for Planned Activities**	**(Item 3.2.3)** **Results Available** **- Yes?** **- No?**	**(Item 3.3.1)** **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 3.3.2)** **Schedule Or Milestones for Planned Activities**	**(Item 3.3.3)** **Conclusion Available** **- Yes?** **- No?**
of MCR against radiation”						
Temelín NPP – ensure habitability of MCR and ECR after transition of a severe accident into the ex-vessel phase	Planned	Medium-term 2018	No	Planned Regulatory activity	Medium-term 2018	No
Training management in severe accident scenarios (Organizational measures – training TSC)	Ongoing	Periodic annually	Yes	Ongoing Regulatory activity	Periodic annually	Yes
Analyses of usability of the equipment for SAMG	Ongoing	Short-term 2013	Yes	Ongoing Regulatory activity	Short-term 2013	Yes

4. NATIONAL ORGANIZATIONS

4.1 INTRODUCTION

In this part, the activities performed by the Licensee and state regulatory authority in relation to the national organization are described.

The state regulatory authority is State Office for Nuclear Safety (SÚJB), an independent central state administrative body in the field of nuclear safety and radiation protection. As for its powers and competencies, it is subordinated neither to the Ministry of Industry and Trade nor to the Ministry of Environment. The SÚJB budget represents an independent part in the state budget of the Czech Republic approved by Czech Republic's Parliament. SÚJB is headed by the chairperson appointed by the Government of the Czech Republic. Since 1984 SÚJB (sooner ČSKAE) has been submitting regular annual reports on the results of its activity to the Government of the Czech Republic.

The holder of the licence for the commercial plants operation (based on Atomic Act wording) is the shareholding company - ČEZ, a. s., which is the operator of Temelín NPP and Dukovany NPP – i.e. all the nuclear power plants in the CR. ČEZ, a. s., was founded in 1992 by the National Property Fund of the CR. The main shareholder is the Czech Republic, for which Ministry of Finance of the Czech Republic performs the administration of its equity share. The main subject of activity of ČEZ, a. s., is the production and sale of electricity and associated support of electricity system. It also deals with the production, distribution and sale of heat.

4.1.1 Legislative environment

The base of the legal framework for this field is the Atomic Act (Act No. 18/1997 Coll., of the peaceful utilization of nuclear energy and ionizing radiation), approved by the Parliament of the Czech Republic in January 1997. The Atomic Act authorized SÚJB by the execution of the state administration and regulation of nuclear energy utilization and radiation activities and redefined the scope of its powers and competencies.

The Atomic Act defines the conditions for the peaceful utilization of nuclear energy and ionizing radiation including activities for which the approval or permit of SÚJB is necessary. In the extensive enumeration of duties of Licensees, also the duties associated with its preparedness for the occurrence of radiological emergency are stated among others.

The legislative framework is analyzed in the chapter 2 of the National Report of the Czech Republic 2010 and more detailed requirements following from the legal regulations – relevant for the given chapter – are analyzed with more details in chapters 1, 2, 3 and 5 of this Extraordinary National Report of the Czech Republic.

4.2 DESCRIPTION OF ACTIVITIES PERFORMED BY THE LICENSEE

4.2.1 Survey of implemented and planned activities

The Licensee follows actively the preparation and publishing new or amended generally liable legal regulations (Acts, Decrees, Directives), concerning the safety of operated nuclear power plants. It also applies in its practice safety guides of SÚJB and it works with the documents published by IAEA, WENRA and other institutions and organizations.

ČEZ, a. s., as Licensee for nuclear power plants operation, set-up conditions for the long-term operation of the current NPPs and for New Build by the responsible and conservative attitude to safety assurance.

The Licensee co-operates closely with a number of external subjects, i.e. with professional institutions, universities and companies focusing on the research and development of the nuclear-energy technologies, with the suppliers of the equipment important for safety, co-operation with schools preparing the future employees for assuring the professional background as well as safe operation of nuclear power plants.

The Licensee follows and analyses carefully all the information about the events in the NPP Fukushima Daiichi, it works also with the analyses of other subjects (state regulatory bodies, international organizations, association of nuclear power plants operators). In accordance with the standard relationships, the Licensee focused immediately after the accident in NPP Fukushima Daiichi on informing the inhabitants in the surroundings of the operated nuclear power plants through the mayors of surrounding municipalities who were informed competently in this way and spread this knowledge.

The Licensee published to the accident in NPP Fukushima Daiichi several information materials some of which were distributed into all the households in the zones of the emergency planning of Dukovany NPP and Temelín NPP.

The chapter 6 summarizes the overall results of evaluation of the safety and safety reserves of Dukovany NPP and Temelín NPP in the light of NPP Fukushima Daiichi accident as analyzed in details for individual topics in the chapters 1, 2 and 3. This extraordinary safety assessment resulted in a conclusion that no serious weaknesses have been identified requiring urgent adopting of countermeasures.

4.2.2 Further steps of the Licensee

4.2.2.1 Conditions created by the state regulatory authority for the safe operation of nuclear power plants

The Licensee follows actively, through the safety department, the preparation and publishing new or amended generally binding legal regulations and it monitors continuously the legislative activity of the legislative bodies of the CR (Chamber of Deputies and Senate of Parliament of CR – laws; central bodies of state administration - implementing regulations to laws).

Prepared legal regulations as well as valid generally binding legal regulations published in the Collection of Laws relevant for the safety of the NPP are monitored continuously by the department for licensing which is a part of the safety department. This department assuring the legislative support for the safety of nuclear power plants monitors also international contracts published in the Collection of International Contracts. In co-operation with other departments of the Licensee, also EU legislation is monitored continuously, moreover the recommendations of IAEA, OECD/NEA and other international organizations are followed.

The survey of results of monitoring the legislation is located on the portal accessible to all the employees of the nuclear power plants, or to the whole ČEZ, a. s. company and it is updated with the one(1)-week period. Moreover, in monthly intervals, the survey of legislation for the last month is sent by the electronic mail to all the responsible persons within the safety department and also to all the other departments concerned by new or amended regulations. In case of a danger of delay, the licensing department informs the concerned departments operatively by electronic mail.

The liabilities for meeting the requirements in individual safety fields are defined by the internal control regulation. The departments being in accordance with the mentioned regulation liable for the concerned safety field shall incorporate the new or amended requirements of generally binding safety regulations in their control documentation. Through the control documentation, the duties for meeting the respective safety requirements are assigned to concerned departments in ČEZ, a. s., and their performance is checked subsequently.

A detailed information of the responsibility of the Licensee for the safety of the operated nuclear power plants is described in the chapter 4 of the „National Report of the Czech Republic for the purposes of the Convention on Nuclear Safety of May 2010.

4.2.2.2 Conditions created by the Licensee for the safe operation of nuclear power plants

Meeting requirements of the national legislation relevant for the safety of nuclear power plants is considered by the Licensee for the minimum level of assuring their safe operation. The Licensee implements safety measures above the scope of the requirements of the national legislation, if they are economically acceptable, technically feasible and if the safety level is increased hereby considerably. To evaluate these measures from the viewpoint of their contribution to increase of the safety level and for deciding on their implementation, recommendations of renowned international institutions dealing with safety of nuclear power plants, as well as operational experience from other nuclear power plants and the professional support of scientific-research institutions and universities are utilized.

For assuring the coordinated attitude to safety management a safety department was established determining the rules for assuring the safe nuclear power plant operation. This department, common for both nuclear power plants, is not subordinated to nuclear power plant director and for this reason it is organizationally independent on operational departments of power plants. It defines the rules for assuring the safety operation of the nuclear power plant, for assuring the safety of employees, it performs the control of observing the requirements on the safety and protection of health of persons and environment, it evaluates the safety level of nuclear power plant operation, it assures the system of feedback of operational experience in the scope of operated nuclear power plants, as well as in the scope of other operators, it performs the activity of failure commissions and it is a not negligible participant during evaluating and analysis of safety inconsistencies and during suggesting the remedies.

The safety department, in co-operation with other departments of Licensee, applies the principles of safety culture, carries out the evaluation of the safety culture level and suggests the measures for its increase.

The safety department has extensive and clearly defined competences, a.o. it has the authority to stop all the activities which could endanger the safety of the operation and health of persons up to the time of taking corresponding measures.

In the responsibility of the safety department is: management of nuclear safety, radiation protection, emergency preparedness, physical protection, technical safety, safety and occupational health and safety and environment protection.

Below the illustrative survey of internal activities of Licensee is stated which are focused on creation and maintaining convenient internal environment for assuring the safety of nuclear power plants.

Preparation of qualified staff

The system of recruitment, choice, psychological diagnostics and training is applied based on the good experience and positive feedback with the co-operation of individual departments liable for staff preparation.

The target of the Licensee is to motivate school graduates for the study of technical subjects at secondary schools and universities, to co-operate with them during the study and to inform them about the perspectives in the field of power engineering. The next step is to choose talented technically orientated students for the continual and future co-operation. During the recruitment the company co-operates systematically with 46 secondary schools and 13 faculties of technical universities with which the co-operation contracts were concluded. The company organizes for selected students as well as pedagogues of secondary schools 5x a year a three-day internship “Nuclear graduation”, during which the students have the possibility to get information about the operation of nuclear power plants, to get to know the work conditions of the power plant and to deepen the particular co-operation with the company. The condition is the verification of the psychological fitness of students for the preparation and performance of the profession operator of secondary circuit enabling to acquire the company scholarship during the following study at the technical university. The same system is applied by the Licensee with university students and pedagogues for whom the two-week Summer University workshop is organized twice a year.

Moreover, the company offers to students the presentations and forums with experts, competition, practice and excursions into the operation site of the company, consultation of professional student papers/theses. It participates in „open school days“ and job trade fairs and further activities supporting the technical education. The permission holder participates in the creation of learning material or school curriculum, e.g. in co-operation with Vysočina region it initiated the establishment of the new specialization Power Engineering at High Technical School in Třebíč.

The Licensee participates, through its specialists also directly in teaching at universities, which, in view of their professional focus, prepare the students for the possibility to work at the nuclear power plants in the field of nuclear safety, radiation protection and emergency preparedness. These are a.o. University of South Bohemia, Technical University of Ostrava, University of Defence Brno, University of Pardubice, Brno University of Technology, Czech Technical University in Prague.

For the universities, the Licensee assures the lectures focused first of all on the safety aspects of the nuclear power plant operation. The target of the co-operation is to provide to students detailed knowledge of assuring the nuclear safety and radiation protection during the work with the sources of ionizing radiation at nuclear power plants. The students are informed of generally binding legal regulations applicable to the field of radiation protection as well as emergency preparedness in the Czech Republic and of the development of the relevant international recommendations. The attention is devoted also to the issue of the connection between assuring the nuclear and radiation protection with emergency preparedness, classification of abnormal occurrences at nuclear energy equipment, elaborating emergency plans in relation to occurrence of possible radiological incidents and emergencies. The students get acquainted with the requirements on matter-of-fact contents and the scope of radiation monitoring programs at nuclear power plants (surrounding, outlets, workplaces, persons) and organizational as well as technical assurance of radiation protection at nuclear power plants. In the framework of the co-operation with universities, the employees of the Licensee are consultants for a number of thesis and dissertations.

The result of all the activities is to acquire and to support the technically gifted students with the following offer of the job in the company. The remuneration for the systematic work with the schools is the repeated acquisition by ČEZ, a. s of the title The Most Desired Company – Employer of the year, elected by the students.

Operative staff is supplemented especially from the rows of the successful graduates by the above mentioned recruitment activities, with whom the long-term co-operation was established already in the past. Another source is the company's personnel database of the applicant for a job. A usual form is also the selection procedure. Each external applicant shall meet the determined requirements – degree of school education, professional specialization or practice. If he/she meets the given criteria, his/her psychological fitness for the preparation and profession execution shall be verified.

Psychological diagnostics being assured in a uniform way for both nuclear power plants of the Licensee has a comprehensive character and is focused on the evaluation of the performance ability as well as the personality characteristics of applicants. Especially the technical talent, combination thinking abilities, resistance against stress, ability of long-term concentration, emotional stability, conservative attitude to solution of operational situations, orderliness, reliability of deciding, observing the rules, team communications, elimination of dependence on addictives etc. are accentuated. During OSART mission which took place in Dukovany NPP in 2011, the psychological diagnostics of the Licensee was involved into good practice: „The power plant utilizes the unified access to recruitment, selection, psychological evaluation and training of new employees. The result of this attitude is the permanently high level of success during the tests for acquiring the permission of the operator and finding suitable potential candidates for various power plant departments.“

The target of the preparation of the operative staff is deepening and supplementing the knowledge, skills and habits of newly adopted employees acquired up to now, by specific knowledge and practical experience with NPP issues necessary for the independent performance of the respective activity. The professional preparation is carried out in the module system in such a way that theoretical lectures in the classroom are combined with the practical stay in the NPP and training at the simulator suitably. Besides the standard training programs and plans, a number of further training activities focused on practicing the co-operation of various departments and experts utilizing simulators as emergency training, training of members of the technical support center etc. are implemented in NPP. An especially important training from the viewpoint of increasing nuclear safety and training of co-operation with external subjects is training of complete loss of power supply (station black-out) and its renewal.

Analysis of the power plant risks show that the external events, complete loss of power supply of the power plant (SBO) is a dominant factor contributing to the frequency of the important damage to the reactor core. Comprehensive scenarios of the training of reaction on failures were created and implemented; they are performed at the simulator with the participation of the safety technicians of the power plant, network operators, employees of the transmission system, employees of emergency response, control operative staff of the control room, control operative employees in the terrain and training team. The training involves the models of real failures in the electric network, real response time of employees, interventions for the power supply renewal; the time scope of scenarios was extended, to be able to follow the process of deciding of operators, their communication and team work. The scenarios of the training involve moreover the isolated, the so-called „island" operation of the unit, setting the units in the mode of own consumption, complete automatic reserve replacement and complete loss of power supply; they provide in this way opportunities to higher quality of operational procedures of power plant as well as external organizations and their mutual communication.

Management of nuclear power plants life-time

The long-term operation of nuclear power plants is conditioned by the management of the life-time of nuclear power plants in total and management of the lifetime of their individual installations.

Management of lifetime is part of the process „Care for equipment“. The principles, basic principles and access to management of lifetime is defined in the internal management document of the Licensee „Lifetime Management of ČEZ' Nuclear Power Plants“. The life-time is managed for all the equipment, however, a graded access to management of lifetime based on categorization of the equipment is set. The requirements following from the document „Lifetime Management of ČEZ' nuclear power plants“ and associated documentation are gradually introduced in EDU and ETE.

In EDU, introducing the requirements on lifetime management is a part of LTO program (Long Term Operation). To support introducing the requirements for lifetime management, SW application was created and is utilized in which especially the information from Ageing Management Review implementation for the determined group of equipment are saved. Another SW application is created for the operationally-functional stage of lifetime management.

The management of lifetime is performed in accordance with SÚJB requirements and is evaluated during regular annual updating of safety reports for nuclear power plants handed over to SÚJB.

In the field of lifetime management and preparation of the long-term operation of nuclear power plants, the technical support is provided by a number of professional organizations in the Czech Republic (ÚJV Řež, The Institute of Applied Mechanics Brno Vítkovice, Škoda JS /Škoda Nuclear Engineering etc.). An important support is provided of course by the experts from the international environment, e.g.

- IAEA (SALTO) programs,
- IAEA „Peer Review “ and Follow-up missions,
- membership in EPRI – utilization of programs for Ageing a Long Term Operation,
- participation in NULIFE – ACCEPT project (Ageing of Concrete and Civil Structures in Nuclear Power Plants),
- Verlife (evaluation of life-time of operated VVER),
- co-operation with Slovak NPP (NPP Jaslovské Bohunice and NNP Mochovce) and further power plants of type VVER (Loviisa and PAKS), in the scope of VVER club – annual exchange of information in the form of work meetings,
- application of WANO programs.

The achieved level of lifetime management will be evaluated within PSR Dukovany NPP in 2014, for Temelín NPP the years 2018 – 2019 are planned.

Nuclear fuel and reactor physics

Nuclear fuel and reactor physics field is supported by a number of organizations from the research and development preparing for the Licensee a verified and by SÚJB approved calculation apparatus for the design work within the process of licensing new types of fuel, i.e. neutron-physical, thermo-hydraulic codes and codes for the solidity calculation of fuel elements and simulation of fuel behavior.

The Licensee supports the research especially in the field necessary for the design of the fuel assemblies and reactor core, calculations reflecting implemented changes (suppliers Škoda JS, ÚJV Řež), testing of material for fuel cladding (Nuclear Fuel Institute/ÚJP Prague, a.s.), etc.

To support the development of SW means for the creation of safety reports on nuclear power plants, transfer of results and exchange of information in the scope of international work teams and research organizations is supported and assured. The transfer of information and know how is assured through ÚJV Řež (e.g. within OECD Halden Reactor Project, Studsvik Clading Integrity project II, CABRI project – experiments with reactivity, implemented in France etc.).

The Licensee is involved in a number of activities and projects also directly. For example, he takes part in activities of NPP operators group – TUG (The Utility Group), where he participates in the exchange and transfer of information among European NPP operators in the field of design, changes and behavior of nuclear fuel.

Moreover the representatives of the Licensee participate in the activities in the scope of:

- AER group (Atomic Energy Research) - where the specific issued connected with VVER are treated,
- EPRI project – transfer of information relating to fuel operation –etc.

In the scope of preparation of the licensing process of the newly introduced fuel types in the CR, the sub-supplier organizations (RNC KI, OKB Gidropress, VNIINM Botchvara) of the fuel supplier (JSC TVEL) perform the safety analyses being submitted to the State Office for Nuclear Safety during the whole licensing process.

Parallel with analyses submitted by the fuel supplier, the analogical analyses are performed also by domestic organizations (ŠKODA JS a.s., ÚJV Řež a.s., ÚJV Řež – Energoprojekt Praha), applying the same input data, but SW different from the SW of the supplier and its sub-suppliers. A part of the SW utilized by the Czech organizations is developed in domestic laboratories (STAMOD-440, CALOPEA, MOBY DICK). Safety proofs submitted to SÚJB in the Pre-operational Safety Report are based on the analyses of these Czech organizations.

The SW being the product of Czech organizations (ŠKODA JS, ÚJV Řež – Energoprojekt Prague) is utilized for the design of fuel charges and their safety evaluation performed directly by the departments of reactor physics of nuclear power plants.

Analyses of radiological consequences of operational events on the surroundings of Dukovany NPP are performed exclusively by the Czech researchers (Škoda JS a.s.).

SW applied by the organizations of the fuel supplier as well as domestic organizations for the calculations of the thermo-hydraulic as well as neutron-physical characteristics of the fuel assemblies as well as reactor core and safety analyses is subject, based on the requirement of the Atomic Act to the process of evaluation of its suitability for performing analyses and safety analyses important for nuclear safety. In the same way, also the SW representing a part of the system of reactor core monitoring is evaluated.

Detailed information about assuring the nuclear safety, radiation protection, technical safety and emergency preparedness are stated in the National Report of the Czech Republic for the Purposes of the Convention on Nuclear Safety" of May 2010, assuring the emergency preparedness and response and management of post-accident states is described in more detail in the chapter 5 of this „Extraordinary National Report".

Technical safety and metrology

In the field of technical safety, the Licensee has introduced, for performing the non-destructive testing, a two-degrees control carried out by two departments being independent on the direct management of work. The qualification of the main methodologist for non-destructive testing is on the highest level - Level 3 according to EN 473.

The department liable for technical safety management is an accredited body for performing the inspection activities on the NPP equipment most important for safety (i.e. classified equipment with a special design set by the Decree No. 309/2005 Coll.) as the so-called „inspection body type B “. In the regular intervals, it is reaccredited by the national body for accreditation – Czech Accreditation Institute (ČIA).

The program of operational inspections is managed by safety department independent on the management of work. The program of operational inspections is processed in the form of a license document to be approved by SÚJB. The safety department submits to the regulatory body (SÚJB) the results of Program performance.

Each general repair is terminated by meeting of „Expert commission for evaluating the results of operational inspections“, where the performed inspections are assessed and evaluated with the participation of producers, research and professional institutes. Approving record on meeting of the Expert commission is one of the conditions for granting the permit for the start of the unit after the shutdown for the nuclear fuel exchange.

The authorized employees of the safety department represent the Licensee in ENIQ (international organization for inspections and qualification). The most important control activities are qualified in accordance with ENIQ methodology.

To meet the requirements following from the metrological generally binding legal regulations, the Metrological laboratories of NPP are accredited by the national accreditation institute (ČIA) according to ČSN EN ISO/IEC 17025 standard: Evaluating the accordance – General requirements on the capability of testing and calibration laboratories.

Metrological laboratories of NPP in the field of ionizing radiation are authorized by the national authority to performances in the state metrological inspection of gauges and they take part regularly in comparing examinations between the laboratories. The employees in the field of metrological assuring of NPP are certified by the national metrological institute (ČMI) and the professional metrological company (ČMS – Assocation of Metrologists).

4.2.2.3 Preparation of new nuclear source – units 3 and 4 in Temelín NPP site

ČEZ, a. s., has decided based on the evaluation of conditions and energy needs to build two new reactor units. It is planned to situate these units into the current Temelín NPP site. At present, the preparation of new reactor units ETE 3,4 is in the stage of elaboration of the bids for the published inquiry by potential suppliers.

The construction of the nuclear installations is governed by general regulations for the civil structure construction on one hand and by the special legislation concerning peaceful utilization of nuclear energy and ionizing radiation on the other hand.

The general regulation for any kind of construction is the Act No. 183/2006 Coll., Building Act. Based on this Act, each construction is split in the stages: construction sitting, constructional stage, trial operation stage and stage of permanent operation (for the purposes of this report, the description of the procedure under the Building Act and associated regulations is strongly simplified). Start of each of the mentioned stage is conditioned by the issue of the permits under the Building Act (for the civil structure location, it is a Decision on sitting of the structure, for the construction: construction permit, for the permanent operation: final inspection approval (equivalent to the commercial operation license).

The issue of individual permits under the Building Act is conditioned by the issue of the obligatory standpoints of all the concerned state administrative bodies. Without these obligatory standpoints, the respective permission under the Building Act cannot be issued and the respective construction stage cannot be started. If any obligatory standpoint is negative (disables to grant an application for the license under the Building Act), the building authority rejects the application for license issue under the Building Act.

The obligatory standpoints of the State Office for Nuclear Safety (SÚJB) represent principal obligatory standpoints for issue of the decision on location of the structure under the Building Act.

For issue of the decision on siting of the structure under the Building Act as well as for issue of the site approval of the nuclear installation under the Atomic Act the standpoint of the respective body of state administration in the matter of environmental impact assessment is also necessary.

At present, transboundary assessment according the Act No. 100/2001 Coll., on environmental impact assessment is being performed. In accordance with the Atomic Act and its implementing decrees, especially the SÚJB Decree No. 215/1997 Coll., on siting criteria for nuclear installations and SÚJB Decree No. 195/1999 Coll., of requirement on nuclear installation for assuring the nuclear safety, radiation protection and emergency preparedness, the documentation for the application for SÚJB permit for sitting of ETE 3,4 reactor units is being prepared.

Concerning the new nuclear builds it was not necessary due to events in NPP Fukushima to perform immediate modification in the generally binding regulations relevant for the field of nuclear energy.

4.2.2.4 Approach of Licensee to assuring communication under the normal operation

Under the normal operation, the Licensee meets all the requirements of the generally binding regulations concerning the communication or requirements on transmitting the information in the field of safety of nuclear installation (e.g. requirements of Atomic Act and its implementing regulations for transmitting information important for safety to bodies of state administration – especially to SÚJB, process and formal requirements of the administrative order, requirements of the State Inspection Act etc.).

In the control documentation, unambiguous competences of the Licensee for the communication with bodies of state administration and with other interested persons (with SÚJB, municipalities in surroundings of nuclear installation, entities involved into the emergency preparedness system – integrated rescue system parts – policy, firemen, medical service, Local Authorities, …) are defined. Detailed information about assuring the communication under the normal operation and in case of abnormal occurrences with state administration bodies, with the municipalities in the surroundings of nuclear power plants as well as with general public and with entities involved into the treatment of abnormal occurrences are stated in the chapter 11 „National reports of the Czech Republic for purposes of Convention on Nuclear Safety“ of May 2010. Assuring the emergency preparedness and response and management of post-accident state is described with more detail in the chapter 5 of this „Extraordinary National Report“.

Unambiguous rules for transmitting information about events important for safety in the NPP to the respective state administration bodies were defined for the Licensee. The persons authorized by assuring the particular communication were defined as well as respective communication channels with the defined form of transmitted information, terms for transmitting information etc.

Unambiguous rules for standard communication with bodies of state administration in the scope of administrative procedures, for the communication in the scope of control activity of state regulatory authorities, for work negotiations with bodies of state administration, local self-government etc. are set for the Licensee.

Intensive attention is devoted by the Licensee to the communication with inhabitants of the Czech Republic. In both localities of nuclear power plants, information centers are built providing to all the interested persons in an understandable form: information about nuclear power engineering, information about nuclear physics, information about potential of nuclear power engineering in future, about topical problems in utilization of nuclear energy for energetic purposes (accident of NPP Fukushima Daiichi), etc. Excursions into power plant sites are organized for the visitors.

Through the web sites and phone lines the questions concerning the nuclear power engineering are answered (e.g. influence of nuclear power plant on the surrounding environment, treatment of spent nuclear fuel and radioactive waste, emergency preparedness etc.).

For assuring the communication with inhabitants, the communication departments were established at both nuclear power plants. These departments are supported by the professional competence of other departments of the Licensee; they assure the communication with general public. The core of their activity consists in assuring the communication with the public from close surroundings of nuclear power plants (the concern is especially the region limited by zones of emergency planning). They control the activity of information centers, they connect and maintain the personal contacts of power plants representatives with surrounding municipalities (especially of management of nuclear power plants with representatives of municipal self-government – with mayors of municipalities), they provide regular and operative information to public to topical items of nuclear power engineering (publication activity – periodicals, operative information and promotion materials), they organize cultural, sport and social events, they provide financial support of the Licensee to surrounding municipalities, they solve particular problems in relation of the public to the operation of nuclear power plants etc.

4.2.2.5 Activities of the Licensee for creation and maintaining the external environment supporting the safe operation of nuclear power plants

In view of the fact that the safety level of NPP is influenced in a very important way by a number of external subjects, the Licensee devotes attention to co-operation with these subjects, i.e. professional institutions, universities and companies focused on research and development of nuclear energy technologies, co-operation with the suppliers of equipment important for safety, co-operation with schools preparing future employees for assuring the professional background as well as for assuring the safe operation of nuclear power plants.

Within the increase of the safety of operation of nuclear power plants and improving the position of ČEZ, a. s., at the market with electrical energy, ČEZ, a. s., supports in an important way the science and research in the Czech Republic.

In connection with nuclear power engineering, scientific and research projects in the CR as well as within international co-operation are supported. For this purpose, a work group for the science and research was created in ČEZ, a. s,, the task of which is to co-ordinate and to roof all the activities concerning the research and development in ČEZ Group.

The projects supported by ČEZ, a. s. are directed to following fields:

- research of materials,

- development of grids,
- degradation effects of melting on concrete structures etc.

The support is focused also on the co-operation with research institutes and agencies in national well as international projects.

To outline the professional background enabling to the Licensee to assure a safe operation of the nuclear power plants in the Czech Republic, the following survey is stated. This survey focused first of all on the survey of professional background in the Czech Republic is in no case the complete enumeration of the subjects providing the support to the Licensee. Its target is only to illustrate the present state of the experts circle available to the NPP Licensee.

Nuclear Research Institute Řež, a. s.:

- support of safe, reliable and economic operation of nuclear power plants and support of preparation of implementation of the new nuclear source for the Czech Republic;
- design support of nuclear power plants;
- analytic support of the Prevention Program and managing emergency states;
- research and development in reactor physics, fuel cycle, safety analyses, severe accidents, probabilistic safety analysis, emergency preparedness, diagnostics and reliability of the current as well as new reactor technologies;
- technical support in the field of lifetime management and during assurance of the long-term operation;
- prevention and managing emergency conditions;
- chemistry of fuel cycle of the nuclear power plants, treatment of radioactive waste and evaluation of influences of fuel cycles and waste treatment on the man health and environment;
- probabilistic safety analysis (PSA) – maintenance and updating of PSA Level 1 and PSA Level2 models.

TES, s.r.o:

- analytical support of Prevention Program and coping with emergency states;
- design support of nuclear power plants.

Envinet, a.s.:

- analytical support of the Prevention Program and coping with emergency states;
- SW support for measuring the surface temperature of OS Castor.

Research Centre Řež, s.r.o.:

- research and development in the field of nuclear power engineering;
- co-operation with universities during education of new experts for power engineering.

Nuclear Safety & Technology Centre, s.r.o.:

- use of know-how from the construction and operations of NPP - VVER type for the technical support of operators.

The Institute of Applied Mechanics Brno, s.r.o.:

- research and providing services in the field of the machinery engineering and design of the constructions;
- support in the field of management of installation lifetime and in the field of assuring long-term operation;
- design support of nuclear power plants.

Energoprojekt Slovakia, a.s.:

- design and engineering organization working for nuclear power engineering.

Research and Testing Institute Plzeň, s.r.o.:

- research and tests focusing on increase of operational reliability and life-time of energetic installation;
- calculations of solidity, dynamics, damage due to fatigue, deformation resistance and thermo-mechanics;
- mechanical engineering for power plants, metallurgy and material engineering, analytic chemistry, metalography, mechanic testing room, dynamic testing room, noise and vibrations, calibration laboratory.

EGP INVEST, s.r.o.:

- design, engineering, investor and supplier services in nuclear power engineering.

Psychological Institute of Faculty of Philosophy and Arts in Brno, GNOZIS – Association of Psychologists, Jaslovské Bohunice, Slovak Republic:

- psycho-diagnostics.

Universities and High schools (e.g. University of South-Bohemia - Faculty of Health and Social Studies, VŠB-Technical University of Ostrava, University of Defence Brno, University of Pardubice - Faculty of Chemical Technology, University of Technology Brno, Czech Technical University in Prague, High Technical School of Třebíč):

- preparation of experts for the work at nuclear power plants, especially in the fields: nuclear safety, radiation protection, emergency preparedness, reactor physics, care for technical equipment important for safety.

National Institute of Public Health, Laboratory of genetic ecotoxicology of National Institute of Public Health, Institute of Experimental Medicine of Academy of Sciences of CR:

- project „Evaluation of impact of Temelín NPP on environment" – cyto-genetic analysis of peripheral lymphocytes of NPP employees.

Škoda JS, a.s.:

- engineering, production and service of components for nuclear power plants;
- safety analyses for nuclear power plants;
- manufacturing of facility for storage of spent nuclear fuel (containers CASTOR are supplied not only for NPP in the Czech Republic, but they are also exported abroad – Lithuania, Bulgaria); Škoda JS, a.s. participated in the development and improving of many technologies necessary for the production of containers;
- support in management of installation lifetime and in the field of assuring the long-term operation;
- development and application of neutron-physis and thermo-hydraulic codes for the design of fuel assemblies and reactor core;
- design support of nuclear power plants.

VÚJE, a.s., Trnava:

- processing RTARC program - analytic support of safety evaluation.

VÚJE Česká republika, s.r.o.:

- preparation, implementation and coordination of work during putting NPP in operation;
- analytic support of Prevention Program and coping with emergency;
- solving tasks from the field of nuclear power plants operation.

ČEZ ENERGOSERVIS, s.r.o.:

- assembly, repairs, revision and tests of NPP facilities (especially equipment of primary part);
- treatment of dangerous waste.

ÚJP, a.s., Praha:

- testing material for fuel cladding in the normal and postulated emergency conditions.

I&C ENERGO, a.s.:

- design support of nuclear power plants.

Královopolská RIA, a.s.:

- design support of nuclear power plants.

Královopolská SAG, s.r.o.:

- design support of nuclear power plants.

Meacont Praha, spol. s.r.o.:

- design support of nuclear power plants.

Škoda Power, s.r.o.:

- design support of nuclear power plants.

ZAT, a.s., Příbram:

- design support of nuclear power plants.

Vítkovice, a.s.:

- supplier of important components for NPP facilities.

POLDI Hütte, s.r.o. Kladno:

- supplier of important components for NPP facilities.

To assure and to maintain a high NPP safety level, the Licensee co-operates with the renowned suppliers of the installations for nuclear power plants. The co-operation with foreign companies and organizations which participated in elaborating the design of nuclear power plants is of high importance, of which it is possible to name e.g. the companies: Westinghouse, OKB Gidropress, TVEL.

4.2.2.6 Reaction of Licensee on accident of NPP Fukushima Daiichi

The Licensee devotes intensive attention to co-operation with the inhabitants of the surroundings of nuclear power plants, as the positive relationship of surrounding municipalities to the nuclear power plant is the necessary condition for the long-term trouble-free operation of NPP. During the co-operation with inhabitants, providing objective and understandable information about the operation of nuclear power plants, their influence on the surrounding environment and the public education and preparation for cases of potential crisis situation is accentuated. Information about events on NPP Fukushima Daiichi are monitored and analyzed carefully by the Licensee, he works also with analyses of other subjects (state regulatory authorities, international organization, association of operators of nuclear power plants, etc.).

A more detailed information about the communication with inhabitants in the scope of emergency preparedness is stated in the chapter 11 „National report of the Czech Republic for the purpose of the Convention on Nuclear Safety“ of May 2010 and in chapter 5 „Emergency preparedness and emergency response“ of this Extraordinary National Report.

In Dukovany NPP and Temelín NPP the answers on questions of the public concerning the safety are provided by the respective communication departments. For providing information to wider public, e.g. information material is utilized as well as very good co-operation with surrounding municipalities, first of all with mayors of these municipalities.

In the time of escalation of accident in NPP Fukushima Daiichi the communication departments provided numerous information of cause and course of the nuclear accident in the NPP Fukushima Daiichi to inhabitants. The communication departments of Dukovany NPP and Temelín NPP focus on inhabitants in the surroundings of these nuclear power plants, but in the event in NPP Fukushima Daiichi the communication departments co-operated also with central departments of ČEZ, a. s. company and provided the support for the communication of the Licensee in the nation-wide scope. The increased number of phone, e-mail as well as personal questions especially in the first day after the accident did not require taking any special measure above the standard coping with questions.

In accordance with the standard relationships, the Licensee focused immediately after the accident in NPP Fukushima Daiichi on informing inhabitants in the surroundings of Dukovany NPP and Temelín NPP, utilizing very good relationships with the representatives of local government (mayors) of surrounding municipalities. Through the mayors being competently informed by the Licensee, this information arrived to particular questioners.

Already in the first day after the accident of NPP Fukushima Daiichi, a meeting with the group of mayors of municipalities from NPP surroundings took place in the framework of the long-term co-operation focused on informing inhabitants of assuring the power plant safety. This group of mayors is trained in the nuclear power plant issues and in accordance with the regulations of physical protection they have open access to the power plants. The mayors are provided with detailed information about nuclear installation operation in order to be able to answer questions of citizens. The meeting focused on providing information available about the accident at that time. They were also informed continuously about the development of situation in Japan through regular reports sent by the communication department. For example in April and October 2011 meetings with mayors of surrounding municipalities from the whole emergency planning zone of Dukovany NPP (from the circle of 20 km around NPP) took place focused on informing about NPP Fukushima Daiichi.

Informing inhabitants about the events in NPP Fukushima Daiichi took place in similar way in Temelín NPP. Based on the setup information relationships, the information about the accident in NPP Fukushima Daiichi was processed and sent by electronic mail to mayors of all the municipalities in the emergency planning zone of Temelín NPP (circle of 13 km around NPP) already in the middle of March. In the middle of April and subsequently in October seminars on the events in NPP Fukushima Daiichi and to the robustness of the Temelín NPP against extreme natural influences was organized in Temelín NPP for mayors and representatives of the self-government of municipalities.

Detailed information about evaluating the experience and lessons learned from copying with the accident in NPP Fukushima Daiichi are included in remaining chapters of this „Extraordinary National Report“.

4.2.3 Conclusion of the Licensee

To increase the safety level and for correct meeting the requirements of the generally binding legal regulations for safety, SÚJB recommendations issued in the form of safety guides are very important for the Licensee.

The safety guides of SÚJB are in many cases also the information about the direction in which, with high probability, also the generally binding national legislation will go. The Licensee works similarly as with safety guides of SÚJB also with documents issued by IAEA, WENRA and further institutions and organizations.

In the national report „Stress tests“, the results of evaluation of safety and safety reserves of NPP Dukovany and Temelín NPP are interpreted in the light of accident of Fukushima Daiichi NPP. It follows from the stated results that no problems were found out during the extraordinary evaluation of the safety level of the nuclear power plants, requiring taking immediate measures for their removal.

The work at the permanent improvement of the present state concerning the above mentioned fields is done permanently. The Licensee reacts continuously on changes of national legislation and analyses international recommendations; in a number of cases it accepts the recommendations (in co-operation with SÚJB) before their introduction into the binding national legislation.

Detailed information about the safety evaluation of Dukovany NPP and Temelín NPP in the light of experiences acquired during copying with NPP Fukushima Daiichi accident are contained in remaining chapters of this „Extraordinary National Report“, and above all in the Czech National Report on „Stress tests“.

4.3 DESCRIPTION OF ACTIVITIES PERFORMED BY THE STATE REGULATORY AUTHORITY

4.3.1 Survey of implemented and planned activities

4.3.1.1 Activities in legal environment

SÚJB performs, during its activity, continual evaluation of effectiveness of legal regulations concerning the field of supervision of nuclear safety and it suggests changes for this purpose. Besides the process of preparation of the new Atomic Act, important amendments of the current Atomic Act are mentioned below.

Amending of the Current Atomic Act (Act No. 18/1997 Coll.)

In 2011 an amendment was made to the new Atomic Law which has introduced a new financing policy of SÚJB. By this amendment specified fees for professional activities of the State Office for Nuclear Safety have been set up for Licensees or applicants for certain regulatory approval/permits. These fees are paid either once in parallel with submitting an application for the given regulatory approval/permit or regularly (annually) as a maintenance fee during the period of the validity of an issued license.

These administrative fees represent a contribution for the payment of SÚJB cost associated with the issue of such permits and with performance of the state supervision over the activity of Licensee. It is estimated that the fees in the suggested amount cover up to 60 percent of the planned budget of State Office for Nuclear Safety. In case of permitting a construction of an important nuclear installation, however, this share could raise up to more than 70 percent.

Preparation of New Atomic Act

The Preparation of the new Atomic Act was not primarily initiated by the necessity to set completely new legal relationship, but moreover to amend and especially to specify the current legal regulation based on the experience acquired from fifteen years of application of the Atomic Act (and associated legislation) using the above mentioned new recommendations of international institutions and other new process as well as professional knowledge. Also authorizing provisions of the current Atomic Act for the creation of the implementing legal regulations shall be subject to a thorough modification.

Regarding radiation accidents the main reason why to consider the new regulation is the need to bring this field into accordance with the general crisis legislation which gradually came into being in the period after the current Atomic Act came into force in the year 1997. Moreover it is necessary to take into consideration international requirements and recommendations which became stricter in the last 15 years or were at least specified. Last but not least it is necessary to utilize the hitherto experience acquired in this field during the emergency training during which a certain gap for improving the current legal regulation was established. It is necessary to define clearly the liability for individual acts in the system of emergency preparedness and response.

In 2011 the matter-of-fact intention of the Atomic Act commented in the scope of the state administration as well as by professional and general public was finished. At present, it is worked at the wording of the Act as well as implementing decrees.

Issuing safety guides of SÚJB

SÚJB, within its powers and competencies, issues in accordance with the principles of activity of administrative bodies and international practice the safety guides continuously in which it continues to specify the nuclear safety requirements. E.g. in 2010 it completed the series of safety guides including the requirements of WENRA Reference Levels.

Before the issue, each safety guides is submitted for comments to professional public and Licensee.

4.3.1.2 System of training of SÚJB inspectors

The professional preparation of employees and maintaining their qualification is assured in accordance with the internal SÚJB guideline. The base is the so-called individual plan of personal growth of the employee, which is evaluated and specified regularly. The whole process of professional preparation is combination of general and specialized training of all the employees without difference in position or kind of executed activity.

Within the inspectors training, special courses focused on nuclear technologies and training at full-scope simulators of control systems of the nuclear power plants are repeatedly organized based on the business contract, in the training center of the Licensee ČEZ, a. s. in Brno. Inspectors also take part in the internal SÚJB seminars organized to every important or event interesting from viewpoint of SÚJB activity. The program of the seminars is focused especially on the description of abnormal occurrences and analysis of their causes.

In co-operation with ČVUT/Czech Technical University (Faculty of Nuclear Sciences and Physical Engineering), in 2010-2011, the training of internal SÚJB lecturers took place. The trained lecturers - SÚJB inspectors will provide training of newly hired employees and deepening on the qualification also of the current ones.

4.3.1.3 External support

SÚJB utilizes the support of organizations in its branch. These are e.g. State Institute for Radiation Protection (SÚRO), which is a public research institution providing the professional and technical support for SÚJB in the field of radiation protection and the public research institution The National Institute for Nuclear, Chemical and Biological Protection (SÚJCHBO), providing the primarily professional and technical support of SÚJB.

Concerning the direct external support outside its branch, SÚJB co-operates with a number of technical organizations. A very intensive technical support is provided especially by the Research Centre Rez, s.r.o. (TSO Section for SÚJB support). This support concerns especially the expert evaluation of the safety analyses submitted to SÚJB by the Licensee of Dukovany NPP and Temelín NPP in safety analysis reports.

In the field of inspectors education it is e.g. the above mentioned co-operation with ČVUT.

In connection with the expected application of the Licensee for the site approval of the new nuclear builds, a number of activities of the technical support focused on the issue of site evaluation. E.g. the project solved by the Institute of Geology of the Academy of Science of the Czech Republic is being completed and the output is the recommendation for SÚJB inspectors for evaluating the respective parts of safety documentation and elaborating the base for the issue of the safety guide for evaluation of sites. With the support of Enconet Austria, the comparison of the national requirements and criteria for NPP siting with topical safety recommendations of IAEA was performed.

4.3.1.4 Communication with external environment

In connection with the event in NPP Fukushima Daiichi SÚJB, particularly SÚRO communicated with CR' Government, Embassy of CR in Japan, media, general public, relevant ministries and their subordinated organization and last but not least also with international organizations.

Similarly like in other countries operating nuclear power plants, also in the Czech Republic the activities were initiated directed on the evaluation of the nuclear safety level in relation to this accident.

SÚJB appointed immediately a group of experts with the target to evaluate the situation based on the analyses of reports received from Japan and subsequently to assure the communication and objective informing of general public of the situation and to maintain in parallel the working communication with representatives of the owner/Licensee for the preparation of reports and information required by the European Commission after the accident.

In view of the scope of work and providing independent analyses, SÚJB called in the analytic expert support from the Research Centre Řež, s.r.o. (hereinafter referred to as CVŘ) and in the field of radiation protection the analytical group of selected employees of the National Radiation Protection Institute SÚRO. In co-operation with CVŘ the websites were put in operation where the citizens had the possibility to ask questions in connection with the event at Japanese Fukushima Daiichi NPP.

Analytical group of SÚRO prepared in the first days the report on the topical situation in Japan, reactions and standpoints in the world and on the topical radiation situation on our territory with the frequency 2x a day. After the situation stabilized the frequency of transmitting reports decreased to 2x a week.

SÚJB established in cooperation with Research Center Řež, s.r.o. a special web portal (SÚJB is its coordinators), through which the questions of general public concerning the accident of NPP Fukushima Daiichi are accepted.

In connection with stress tests of the Czech nuclear power plants SÚJB created a special section also on its web site in which it brought to the Czech general public the most topical information about individual stages and topical results of safety inspections of both power plants. After publishing the final report it also asked the general public for comments in the interest of increased transparency of the whole process.

4.3.2 Further steps of the state regulatory authority

As described in this part 4.3 of this Extraordinary National Report, individual activities for assuring the nuclear safety of operated nuclear installation are long-term and conceptual from the side of SÚJB as state regulatory body.

An important impulse is the construction of two nuclear units in Temelín NPP site planned by ČEZ, a. s. For this purpose SÚJB strengthens its network of technical support organizations and in view of the planned recruitment of new inspectors (for licensing these new nuclear units as well as substitution for the employees intending to retire) it introduces a more effective system of internal education of inspectors.

A direct response on the events after Fukushima Daiichi is a more intensive communication with the external surroundings, especially with the general public. In its scope, the scale of current communication means was extended e.g. by newly created www site and discussion forums, the further usage will depend on the interest of public.

4.3.3 Conclusions by the state regulatory authority

As described above, it was not necessary initiating qualitatively new activities in the fields stated in this chapter on the part of the Licensee. Similarly, the activities of SÚJB do not deviate from the usual scope being performed in the framework of its defined powers and jurisdictions.

4.4 FINAL SUMMARY OF CHAPTER 4

<table>
<tr><th rowspan="2">Activity</th><th colspan="3">Activities by the Licensee</th><th colspan="3">Activities by the State Regulatory Authority</th></tr>
<tr><th>(Item 4.2.1)

Activity

- Taken?
- Ongoing?
- Planned?</th><th>(Item 4.2.2)

Schedule
Or
Milestones
for Planned
Activities</th><th>(Item 4.2.3)

Results
Available

- Yes?
- No?</th><th>(Item 4.3.1)

Activity

- Taken?
- Ongoing?
- Planned?</th><th>(Item 4.3.2)

Schedule
Or
Milestones
for Planned
Activities</th><th>(Item 4.3.3)

Conclusion
Available

- Yes?
- No?</th></tr>
<tr><td colspan="7">Topic 4
National Organizations</td></tr>
<tr><td>Preparation of qualified staff/ exchange of generations</td><td>Is in progress</td><td>Long-term</td><td>Yes</td><td>Is in progress Supervisory activity including issue of safety guide</td><td>Continuously</td><td>Yes</td></tr>
<tr><td>Management of lifetime / LTO Dukovany NPP</td><td>Is in progress</td><td>Long-term</td><td>Yes</td><td>Is in progress Supervisory activity</td><td>Continuously</td><td>Yes</td></tr>
<tr><td>Communication with external environment</td><td>Is in progress</td><td>Long-term</td><td>Yes</td><td>Is in progress</td><td>Long-term</td><td>Yes</td></tr>
<tr><td>Preparation of new Atomic Act</td><td>x</td><td>x</td><td>x</td><td>Is in progress</td><td>2012/2013</td><td>Partial Yes</td></tr>
<tr><td>PSR performance</td><td>Is in progress</td><td>Long-term</td><td>Yes</td><td>Is in progress Supervisory activity including issue of safety guide</td><td>Continuously</td><td>Yes</td></tr>
</table>

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5. EMERGENCY PREPAREDNESS AND EMERGENCY RESPONSE

5.1 INTRODUCTION

In the introduction of this chapter, legislative environment in the field of the internal as well as external emergency preparedness including the information of the last important amendments is shortly described.

Moreover, the organization of emergency response on the part of the Licensee, the way of the classification and type of events, way of announcing the occurrence of the extraordinary event and way of organization of external part of emergency preparedness is analyzed.

5.1.1 Legislative environment in the field of the internal and external emergency preparedness

The national legislation is in accordance with IAEA documents, as e.g. TECDOC 718, „A Model National Emergency Response Plan for Radiological Accidents“; TECDOC 953, „Method for Developing Arrangements for Response to a Nuclear or Radiological Emergency“; TECDOC 955, „Generic Assessment Procedures for Determining Protective Actions During a Reactor Accident“.

Legislative framework for the field of emergency preparedness of nuclear installation and their surrounding is represented by the Atomic Act, its implementing regulations and associated government decrees (see the chapter 2.1.2).

Provisions of Section 2 of the Atomic Act define basic concepts – emergency preparedness, radiological incident, radiological emergency, radiological extraordinary situation, emergency radiation, emergency planning zone and emergency plan.

SÚJB, as per Section 3 of Atomic Act, in the scope of its powers:

- approves the on-site emergency plans and their changes after discussing the relationships to external emergency plans; approval of the on-site emergency plan is the condition for the permit to start commissioning of the nuclear facility and operate it,
- determines the zone of emergency planning based on the request of the Licensee,
- controls the activity of the national-wide radiation monitoring network and assures the function of its headquarters,
- arranges the activity of the crisis co-ordination center and the international exchange of data on radiation situation,
- it assures, by a nation-wide monitoring radiation network and based on the evaluation of the radiation situation base decision making documents on measures leading to decrease or averting the irradiation in case of radiation accident,
- is obliged, in appropriate extent, to provide to the public the information of results of its activity, if they are not subject of the state, service or business secret and once a year to elaborate the report on its activity and to submit it to the Government and to the public.

In Section 4 the Atomic Act defines e.g. the principles for performing radiation activities and limiting the emergency irradiation. The principles for averting and decrease of radiation during the radiological emergencies and irradiation of persons participating in interventions are specified in the implementing regulation of SÚJB No. 307/2002 Coll., on Radiation Protection.

In Section 17, the Atomic Act assigns, among general duties, to Licensee the duty to assure the emergency preparedness including its verification in the corresponding scope for individual licenses, and to announce to SÚJB every change important from the viewpoint of emergency preparedness, including the change of all the facts decisive for issuing the license.

Provisions of Section 18 of Atomic Act set, among others, following duties of the Licensee:

- to follow, measure, evaluate, verify and record the quantities, parameters and facts important for emergency preparedness in the scope determined by implementing regulations,
- to keep and to maintain the registration files of sources of ionizing radiation, buildings, materials, activities, quantities and parameters and other facts important from the viewpoint of emergency preparedness and to submit the registered data to SÚJB in the way determined in implementing regulation,
- to perform a systematic supervision of observing the emergency preparedness including its verification.

The provisions of Section 19 of Atomic Act define, as duties of the Licensee in case of occurrence of radiation accident in the scope and way determined by the internal emergency plan approved by SÚJB:

- to inform immediately the respective bodies of public administration, SÚJB and further concerned bodies stated in the internal emergency plan of the occurrence or suspect of occurrence of radiation accident,
- immediately, in case of radiation accident occurrence, to warn the inhabitants in the emergency planning zone,
- to assure immediately the mitigation of consequences of radiation accident in the space where it performs its activity and to implement measures for the protection of employees and other persons from the effects of ionizing radiation,
- to assure the monitoring of irradiation of employees and other persons and leakage of radio-nuclides and ionizing radiation into the environment,
- to inform the concerned bodies especially of the results of its monitoring of the real and expected development of situation, measures taken for the protection of employees and inhabitants, measures taken for management of radiation accident and of real and expected irradiation of persons,
- to check and to regulate the irradiation of employees and persons participating in the management of radiation accident in the space where it performs its activity,
- to co-operate during the mitigation of the consequences of the radiological incident of its installation,
- to participate, in case of radiation accident occurrence in activity of the nation-wide monitoring network.

This section determines moreover the duty of the Licensee to transmit to the respective regional authority and to concerned municipal authorities of municipalities with extended powers the base documents for elaborating the external emergency plan and to co-operate with it at assuring the emergency preparedness in the emergency planning zone.

Moreover it is fixed here that the government order sets the financial share of the Licensee in assuring the activity of the nation-wide radiation monitoring network, antidotes for inhabitants in the emergency planning zone of the respective facilities or workplaces, organizing the press and information campaign for assuring the preparedness of inhabitants in case of radiation accident, in assuring the system of informing the concerned bodies in the scope and in the way determined by the on-site emergency plan, providing the system of warning of inhabitants in their surroundings and the duty of the Licensee to participate in the mitigation of consequences of radiation accident in the emergency planning zone.

Based on the provisions of Section 46 of Atomic Act, some ministries were assigned the duty to participate in assuring emergency preparedness for the needs of radiation monitoring network on the territory of the Czech Republic, particularly:

- Ministry of Finances assures the operation of determined parts of measuring points on border crossings and it participates in assuring mobile groups,
- Ministry of Defence participates in building the network of in-time establishing the radiation situation, measuring points on closures and border crossings, mobile groups and aviation group and it assures the aviation survey means,
- Ministry of Interior participates in organizing mobile groups,
- Ministry of Agriculture participates in assuring the measuring points of water contamination and measuring points of food contamination,
- Ministry of Environment provides meteorological services and participates in functioning the network of in-time establishing the radiation situation, measuring points of air contamination and measuring points of water contamination,
- Ministry of Interior provides the system of information of warning during the assurance of emergency preparedness and its verification.

Moreover it sets that the Ministry of Health shall establish the system of providing special medical aid by selected clinical workplaces to persons irradiated during radiation accidents.

Details and requirements in the field of emergency preparedness for the case of occurrence of extraordinary events (radiological incidents and emergencies) are set by implementing regulations to the Atomic Act:

- SÚJB Decree No. 318/2002 Coll., of details for assuring emergency preparedness of nuclear installations and facilities with sources of ionizing radiation and of requirements on the contents of on-site emergency plan and emergency rules, as amended by SÚJB Decree No. 2/2004 Coll.,
- SÚJB Decree No. 307/2002 Coll., of Radiation Protection, as amended by the SÚJB Decree No. 499/2005 Coll.,
- SÚJB Decree No. 319/2002 Coll., of Function and Organization of Nation-wide Radiation Monitoring Network as amended by SÚJB Decree No. 27/2006 Coll.

SÚJB Decree No. 318/2002 Coll. determines the details for assuring the emergency preparedness of nuclear installations, especially:

- identification of the occurrence of extraordinary event,
- evaluating the seriousness of the extraordinary event and its split into three basic categories,
- declaring extraordinary event,
- activation of intervening persons,
- management of execution of intervention,
- requirements on the intervention procedures and instructions,
- requirements on the program of monitoring the radiation situation,
- way of limiting the irradiation of employees and other persons,
- principles for assuring medical care

- assuring documenting activities in case of extraordinary event,
- transmitting the data to SÚJB of occurrence and course of extraordinary event,
- requirements on preparation of employees and persons,
- requirements on verifying the emergency preparedness and involving the emergency drill and verifying the function of technical means, systems and appliances necessary for management and performing interventions,
- requirements on contents of on-site emergency plan,
- requirements on further documentation for assuring emergency preparedness.

SÚJB Decree No. 307/2002 Coll. states in the provision of Section 92 the general rules for the preparation and performing the interventions and in the provisions of Sections 98 to 100 and in the Annex No. 8 it determines the details for the way and scope of assuring the radiation protection during interventions for decreasing the irradiation in consequence of radiological emergencies. Moreover it determines the initiation values for immediate and subsequent protective measures.

The Government Decree No. 11/1999 Coll. assigns to Licensee the following duties:

- elaboration of the proposal for determining the emergency planning zone of nuclear installation or facility with a very important source of ionizing radiation (the Licensee submits this proposal as per Section 17 of Atomic Act to SÚJB for determining the size of emergency planning zone),
- assuring the activity of nation-wide radiation monitoring network in the emergency planning zone,
- assuring anti-dots for inhabitants in the emergency planning zone
- organizing the press and information campaign for inhabitants in the emergency planning zone for cases of radiological emergencies,
- assuring the system of informing the concerned bodies of occurrence or suspect of occurrence of radiological emergency,
- assuring the system of inhabitants warning in emergency planning zone.

Additional requirements are fixed by the Act No. 239/2000 Coll., of Integrated Rescue System and Change of some Acts, as amended and by the Act No. 240/2000 Coll. on Crisis Management and Change of some Acts (Crisis Act), as amended.

The Act No. 239/2000 Coll. of Integrated Rescue System determines:

- general definition of the extraordinary event being not identical (but wider) in comparison with the concept „radiological extraordinary event“,
- integrated rescue system as coordinated procedure of its parts during the preparation for the extraordinary event and during the performance of the rescue and liquidation work,
- way of management and co-ordination of activity of basic and other parts of the integrated rescue system during the rescue and liquidation work, their co-ordination in place of the intervention by the intervention commander, operational co-ordination and strategic co-ordination by the state bodies, bodies of regions and municipalities with extended authority,
- authorizations and duties of bodies and representatives of regions, municipalities with extended authority and municipalities in case of an extraordinary event on the territory in their territorial authority including the authorization to require the aid from higher bodies and integrated rescue system components,
- rights and duties of the legal entities and natural persons during the preparation for the extraordinary events and during the rescue and liquidation work and protection of inhabitants during extraordinary events including the radiological emergencies,

- split of responsibility and tasks between the bodies of the region, bodies of municipalities with extended authority, municipalities and firefighting and rescue forces of regions during preparing the base documents, elaborating and approving the off-site emergency plans for performing the rescue and liquidation work and inhabitants protection for the zones of emergency planning of nuclear installations and buildings and facilities with dangerous substances.

Act No. 240/2000 Coll., Crisis Act, determines the sphere of activity and authority of state bodies and bodies of territorial self-government units and rights and duties of legal entities and natural persons during the preparation for crisis situation not connected with assuring the protection of the Czech Republic from the external attack, and during coping with them and protection of critical infrastructure; it sets sanctions for breaching these duties.

Implementing legal regulations were published to above mentioned Acts relating among others to assuring the emergency preparedness and the crisis management in the field of usage of nuclear energy and ionizing radiation. The respective details are regulated by:

- decree of the Ministry of Interior No. 328/2001 Coll., of some details of assuring the integrated rescue system, as amended,
- decree of the Ministry of Interior No. 380/2002 Coll., of preparation and performing the tasks for inhabitants protection,
- government Decree No. 462/2000 Coll., for implementing Section 27 para 8 and Section 28 para 4 of the Act No. 240/2000 Coll. as amended,
- government Decree No. 432/2010 Coll., of criteria for determining the element of critical infrastructure.

The Decree of the Ministry of Interior No. 328/2001 Coll., as amended by the Decree No. 429/2003 Coll., sets the details for assuring the integrated rescue system involving the principles of co-ordination and co-operation of its parts/components during a common intervention. Moreover it sets the requirements on the contents of the documentation of the integrated rescue system, way of elaborating the documentation and details of alarm degrees of the alarm plan. The decree determines also the principles and way of elaboration, approving and usage of the emergency plan of the region and off-site emergency plan and principles for crisis communication and connection in the integrated rescue system.

Off-site emergency plan which is emergency plan elaborated for the emergency planning zone is split to:

- information part,
- operative part,
- plans of particular activities.

Information part includes:

a) general characterization of nuclear facility or facilities of category IV,
b) characteristics of the territory, especially as for demographic, geographic and climatic aspects and description of infrastructure on the territory,
c) list of municipalities including the survey of the number of inhabitants and list of legal entities and natural persons doing business which are involved into the off-site emergency plan,
d) results of analyses of possible radiation accidents and radiological consequences for inhabitants, animals and environment
e) system of radiological emergencies classification as per on-site emergency plan,

f) requirements on protection of inhabitants and environment in relation to levels of intervention during the radiological emergency,

g) description of the structure of organization of emergency preparedness in emergency planning zone, including powers of its components for performing necessary activities,

h) description of the system of notification and warning including the relation to the Licensee and transmitting information in the frame of organization of emergency preparedness in the emergency planning zone.

Operative part involves:

i) tasks of administrative authorities, municipalities and components involved in measures from the off-site emergency plan,

j) way of co-ordination of coping with radiological emergency,

k) criteria for declaring the corresponding crisis states if the off-site emergency plan is clearly not sufficient for coping with radiological emergency,

l) way of assuring information flow during the management of mitigation of radiological emergency consequences,

m) principles of activity during extending or possibility of extending the consequences of the radiological emergency outside the emergency planning zone; and co-operation of administrative authorities and municipalities involved in measures from the off-site emergency plan.

Plans of particular activities fix the procedures for performing individual measures, for the fields:

a) information transmission,

b) warning of inhabitants,

c) rescue and liquidation work,

d) sheltering inhabitants,

e) iodine prophylaxis,

f) evacuation of persons,

g) individual persons protection,

h) decontamination,

i) monitoring,

j) regulation of movement of persons and vehicles,

k) traumatological plan,

l) emergency plan of veterinary measures,

m) regulation of distribution and consuming food, fodder and water,

n) measures in case of death of persons in polluted region,

o) assuring the public health and safety,

p) communication with public and mass media.

Decree of MV No. 380/2002 Coll., determines, a.o., the details to the way of informing the legal entities and natural persons about the character of the possible endangerment, prepared measures and the way of their performance for the technical, operational and organizational assurance of the unified system of warning and notification and way of providing the emergency information.

Government Decree No. 462/2000 Coll., as amended by the Government Decree No. 431/2010 Coll. sets especially the details for designating, fixing the mode of filing, manipulation with and archiving documentation and other materials containing special/restricted facts; procedure for clearance of persons for the contact with special/restricted facts; contents and activity of the safety council of the region and determined municipalities and crisis management group of the region and determined municipalities; prerequisites of the crisis plan, plan of crisis preparedness, plan of crisis preparedness of the subject of critical infrastructure and way of their elaboration.

Government Decree No. 432/2010 Coll. determines especially the cross-section and branch criteria for determining the elements of critical infrastructure.

5.1.2 Organization of emergency response (OER) of the Licensee

The system of emergency preparedness (EP) is implemented in accordance with the requirements of legal regulations of the CR and based on IAEA methodology. Assuring EP is one of the basic tasks of the NPPs in the CR. The objective of the EP at NPP is to assure the preparedness of the NPP as well as of the concerned external organization involved in solving EE with accent on:

- Minimizing of the risk of EE occurrence and if EE occurs, on the mitigation of its consequences on-site of NPP and in EPZ,
- Preventing serious health damage during EE.

Strategy of EP is based on the logical development of any event at NPP. EP system, a part of which is also coping with severe accidents, is assured by the complex of measures of the personnel, administrative and technical character. In the personnel field, the objective is creation of OER and assuring the activities related to individual positions in the scope of this organization, in the administrative field the elaboration and implementation of respective procedures, manuals and instructions and in the technical field assuring the functionality and the required scope of the necessary technical means.

The structure of emergency supporting centers is created, from which the OER staff assures the management and performance of interventions. Performing the intervention during extraordinary event occurrence is assured in the first (preventive) stage of event development by the staff of continual shift operation. If the scope of the event exceeds the framework of the possibilities of the staff of continual shift operation, the second stage starts (mitigation of consequences) and standby organization for emergency response is activated (SOER). In this case, emergency response board (ERB) with technical support center (TSC) takes over the responsibility for the management of interventions.

In case of SOER activation, the following emergency support centers are activated: ERB, TSC, external emergency support center, emergency information center and logistic support center. The responsibility for the management of interventions after ERB activation is taken over from shift engineer (SE) by ERB commander.

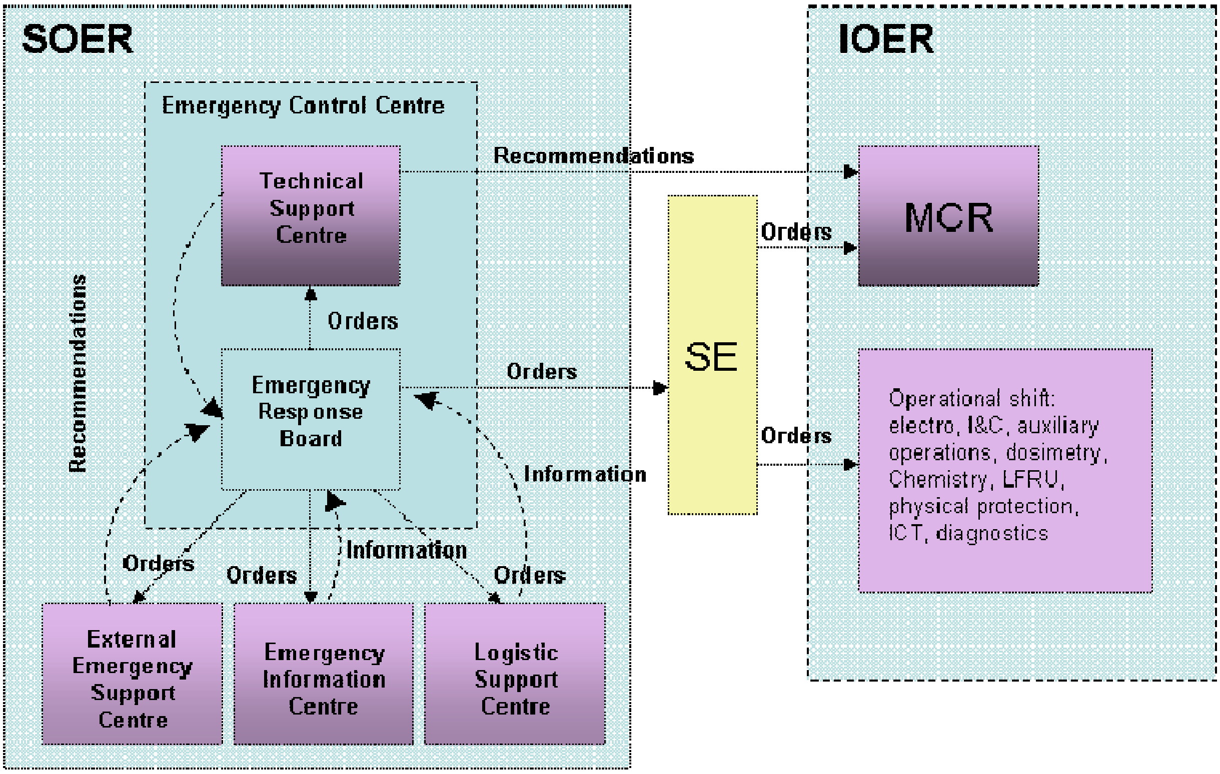

Fig. 1: Structure of OER with stating mutual relationships and information flow

The shift staff performs all the activities pursuant the operational documentation (procedures, instructions, programs…) covering the normal as well as abnormal operation and accident conditions (they include all design basis and partially also beyond-design basis events up to the fuel damage). In all these conditions the shift personnel controls and performs activities with the possibility of support of other technical staff of NPP. In case of occurrence of accident conditions with the fuel damage, the responsibility for the control of activities is transferred to TSC (technical support center) and ERB and the shift staff continues to perform activities as per the requirements of TSC and ERB.

Operative management of the whole NPP is assured by SE.

The SE is liable for performing the classification of EE, declaring EE and performing the activation of the necessary part of OER. If necessary, he/she is authorized to activate a part of OER also sooner than all the criteria for its activation are fulfilled. During the development of EE the SE specifies the EE classification based on the real conditions. After the ERB activation, the ERB commander takes over from SE for the solution and classification of EE.

The management of each NPP unit is in case of EE occurrence assured by personnel of main control room (MCR) and its basic workplace is MCR of the respective unit. If it is uninhabitable, or in case of the loss of possibility of the control of unit technology, this staff assures its activities from the emergency control room (ECR).

Internal organization of emergency response (IOER)

IOHO consists exclusively of the shift personnel, i.e. employees assuring the normal operation of NPP. The staff of the steady shift assures as per SE's instructions all the activities associated with suppressing the symptoms of developing EE up to the time of activation of employees being on duty in the frame of OER.

The SE is, in case of EE occurrence, liable for EE management up to the time when the liability is transferred to the activated ERB's commander. His activity in case of EE occurrence follows the intervention instructions for the SE, where all the liabilities and powers are stated. These are among the most important ones: evaluation of EE seriousness– classification of EE, assuring the notification and warning of NPP staff and warning of inhabitants in EPZ, informing management of NPP and respective bodies and organizations about EE occurrence, decision on SOER activation, and decision on introducing protection measures for NPP staff. The liability for technology remains in SE's authority.

In case of declaring EE depending on the degree of its seriousness, the staff of the steady shift operation (besides the controlling staff of the shift in the control room) continues either: to perform activities as per the respective intervention instructions and instructions of the control staff of the shift, or it meets in a shelter in case of declaring protective measures, from where, it assures based on SE's or ERB's instructions performing the required interventions in the technology or it creates the operative support for LFRS (local fire rescue service) during extricating and rescue work.

To cover the need of taking protective measures of sheltering and evacuation of the staff, shelter teams are made up in order to assure the activation and subsequent operation of shelters at NPP site. The basic duty of members of shelter teams in the shelter is: management of regime in the shelter, registration of sheltered persons, order service, attendance of the air-conditioning, dosimetric measuring of persons, attendance of diesel generators.

Standby organization for emergency response (SOER)

SOER consists of the staff of emergency support centers holding weekly continuous emergency service.

Emergency response board (ERB)

- ERB is the main control body of OER (organizational and emergency response) of NPP. After its activation it assures declaring protective measures for employees and other persons being present at NPP site in the time of occurrence of EE, management of activities of all the employees and other persons participating in performing the intervention during suppressing the development and liquidating the consequences of EE in NPP; it assures the communication with external EP components. ERB assures the deliveries of the necessary material, special means, staff exchange and its material assurance through the logistic support center.

Technical support center

- TSC is staffed with professions in the way enabling to provide qualified technical support to the staff of the control room of the struck unit during EE treatment. TSC staff assures simultaneously the immediate evaluation of the conditions of NPP in view of nuclear safety and radiation protection, it controls the activity of the operatively determined intervention groups during the management of EE consequences and it is able to elaborate the base documents and recommendations for deciding and control activity of ERB. In case of requirement of SE's or ERB's commander, the support for TSC staff by additional specialists may be demanded.

External emergency support centre

- External emergency support center assures the activities associated with the radiation monitoring and evaluation of radiation situation in EPZ and based on the results of radiation monitoring it makes prognosis on further development of radiation situation.

Emergency information center

- The staff of the emergency information center assures, in case of EE occurrence, handing over all the information to mass-media and answering the questions of public. Its activity is focused on informing the lay public as well as bodies of state administration and local government not directly involved into the external emergency preparedness of NPP. It is liable for the preparation of the press reports for the mass media.

Logistic support center

- The staff of the logistic support center provides the necessary material technical means and qualified human resources based on the needs of ERB, TSC and external emergency support center. Logistic support center represents the external support of OER.

5.1.3 Classification of extraordinary events

In case of endangering the safety in the unit or on-site or in case of occurrence of the situation which cannot be coped with by forces of the shift, the shift engineer declares one of the 3 categories of the extraordinary event set as per Section 5 Decree No. 318/2002 Coll., as follows:

- The event of the first degree is an extraordinary event which leads or which may lead to the impermissible irradiation of employees and other persons or impermissible release of the radioactive substances into the space of the nuclear installation or workplace, having a limited local character; for its solution the forces and means of the attendance or shift personnel are sufficient. In case of a transport event, there will be no release of radioactive substances into the environment
- The event of the second degree is an extraordinary event leading or which may lead to the impermissible serious radiation of employees and other persons or impermissible release of the radioactive substances into the environment, not requiring introducing urgent measures for protection of inhabitants and environment. Its solution requires activation of intervening persons of the Licensee and for its management the forces and means of Licensee are sufficient (or forces and means contractually assured by the Licensee)
- The event of the third degree is an extraordinary event leading or which may lead to the inadmissible serious release of radioactive substances into the environment, calling for introducing immediate measures for the protection of inhabitants and environment, determined in the off-site emergency plan and in the emergency plan of the region. The event of the third degree is radiological emergency and its solution requires, besides activation of intervening persons of the Licensee and intervening persons as per the off-site emergency plan or emergency plan of the region, also involving further concerned bodies.

The above mentioned classification corresponds, in principle, to IAEA classification, i.e.:

EE of the 1st degree corresponds to EE classification„Alert“,

EE of the 2nd degree corresponds to EE classification„Site emergency“,

EE of the 3rd degree corresponds to EE classification „General emergency“.

As stated above, OER has been created for coping with EE and has an internal part consisting of the personnel of the shift (IOHO) and emergency part SOER consisting of technical experts of the NPP staff being on duty (within 4 shifts). Readiness of SOER is organizationally assured in such a way that within 20 minutes from the declaration of EE in the working hours and within 1 hour out of the working hours, the respective specialists come to the emergency support centers. The means for activation of SOER staff are backed-up.

Each event important for safety which can lead to EE occurrence if not treated is subject to the assessment of deviations from the normal operation according to the classification system. The classification of seriousness of EE is based on the requirements of the Decree No. 318/2002 Coll. as amended, respecting the recommendations of IAEA in the document TECDOC-955 „Generic assessment procedures for determining protective actions during a reactor accident“. The purpose of EE classification is especially assuring the in-time activation of OER and choice of the suitable and effective response.

The procedure of evaluating the seriousness of the occurred EE in NPP is stated in the respective intervention instructions. Assessing the seriousness of the occurred announced events is performed by SE by comparing the type of the announced event with the pre-defined set of intervention levels. Also ERB' commander is authorized to perform EE classification, if ERB started its activity already and ERB' commander took over from SE the liability for coping with the EE The intervention levels represent the complex of predetermined locally specific initiation conditions; in case of achieving them, NPP state is assessed by the respective classification degree and type. The intervention levels are elaborated for all the operational modes of NPP. Initiation condition may represent exceeding some of the determined parameters, or possibly the occurrence of discrete internal and external events the development of which may endanger nuclear safety and radiation protection at NPP.

In case of declaration of EE of the 1st degree, only the technical part of SOER, i.e. TSC is activated. In case of declaration of EE of the 2nd and 3rd degrees, also the remaining parts are activated - ERB and supporting centers. Up to the time of ERB activation, the activities are controlled by SE and the shift staff shall proceed as per the respective operational procedures.

The workplace of TSC and ERB is Emergency Control Center (ECC), located at NPP site. The organizational way of managing EE is set in the on-site emergency plan approved by SÚJB.

5.1.4 Notification of the extraordinary event

In case of EE occurrence, the immediate reporting of the event to SÚJB, Regional Authority territorially competent for the given NPP, Regional Headquarters of, municipalities with extended powers, technical dispatching of ČEZ, a. s., and to meteo-station in the NPP site shall be carried out. Principal diagram of informing the bodies is shown on the picture. In case of impossibility of establishing the direct connection with SÚJB, the back-up way through Operating and Information Center of General Directorate of FRS CR shall be used which is shown dashed on the picture.

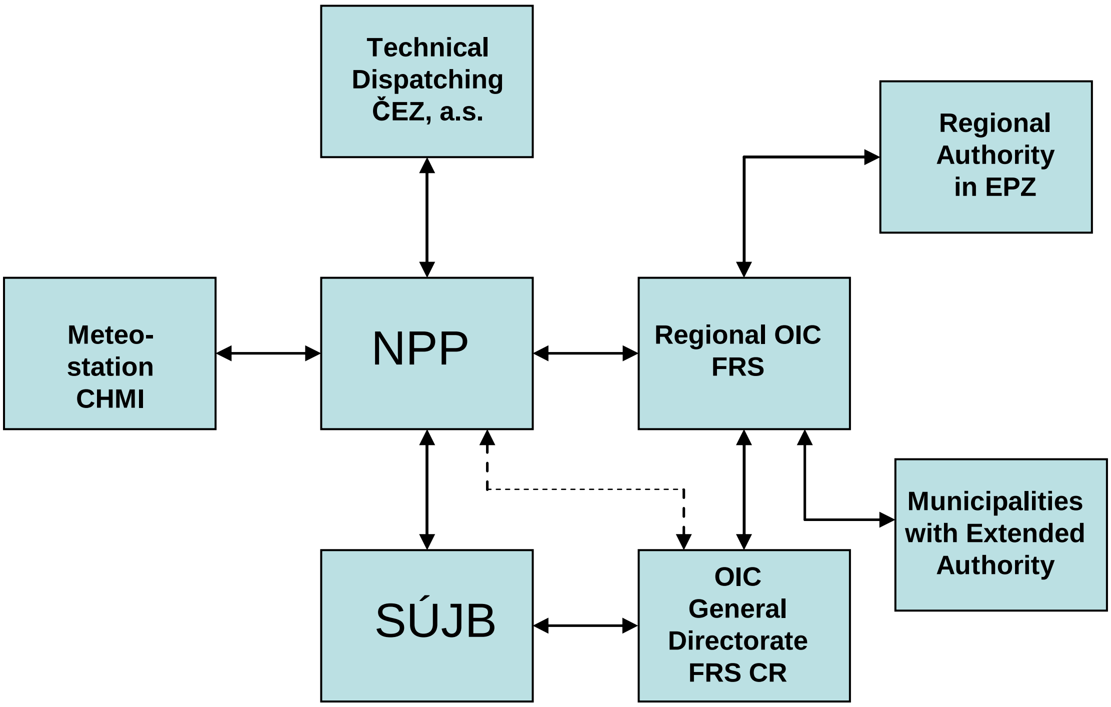

Fig. 2: Informing the external bodies in case of EE occurrence

For necessity of planning of the inhabitants protection assurance in the surroundings of nuclear power plant for the case of radiological emergency and for elaborating the off-site emergency plan, emergency planning zones (EPZ) have been determined by SÚJB decisions (for Temelín NPP it is the territory with the radius of 13 km, for Dukovany NPP the territory with the radius of 20 km). For assuring the measures for the preparation and execution of inhabitants evacuation, the internal part of EPZ has been fixed by these decisions (5km for Temelín NPP, 10 km for Dukovany NPP), as well.

5.1.5 External parts of emergency preparedness

Assuring the external support and possible usage of further capacities, sources and means is managed by the employee executing logistics function in the ERB (emergency response board), in co-operation with the logistic support center.

There is a possibility to acquire aid with the transport or heavy machinery in form of further forces and means through regional OIC FRS locally pertinent to NPP, having the power within IRS to ask further orgnizational components and organizations for material assurance and activities associated with the solution of the occurred EE. Within the whole ČEZ group, the aid for the affected site is assured by the ČEZ crisis board. Within this body, the availability of external specialists would be assured, as well (suppliers, expert knowledge, foreign aid, etc.).

A number of bodies and organizations on the national as well as local level participate in assuring the off-site emergency preparedness of NPP. During the occurrence of EE and the subsequent coping with the occurred EE, NPP communicates with the following external bodies and organizations on the national as well as local level:

SÚJB – Crisis Board

- Crisis Board SÚJB assures, through the radiation monitoring network of the Czech Republic the independent assessment of radiation impacts of the occurred EE. Based on the results of monitoring the radiation situation occurred in the Czech Republic, it provides the base documents for introducing (or cancelling) the measures for the protection of inhabitants as aid for deciding by the crisis board of the respective region.

Regional Authority

- The Regional Authority assures the co-ordination of the off-site emergency preparedness of all the municipalities with extended powers, territory of which is included into the EPZ. The governor (head) of the respective region controls, in co-operation with mayors of the concerned municipalities all the activities connected with assuring the off-site emergency preparedness within the whole EPZ and decides on declaring and implementation of measures for the inhabitants protection. The advisory body for him is the crisis board of the region. Declaring urgent protection measures is performed based on the recommendations of the Crisis Board of SÚJB based on the results of radiation monitoring.

NPP operator (Licensee)

- NPP operator provides, in case of occurrence of radiological emergency at NPP for the crisis board of the region through its emergency response board the necessary co-operation, data and information necessary for evaluating the seriousness of the occurred situation. For assuring the co-operation, NPP sends its representative to the crisis board of the region.

Municipalities with extended authority

- The mayors of the concerned municipalities with extended authority decide on activating the emergency board of their municipalities and control the declaring and implementation of protective measures on the concerned territory of the municipality. Managing these activities is based on the off-site emergency plan. The protective measures are declared each time after the preceding discussion with the emergency board of the region assuring the mutual co-ordination of news and information transmitted among the individual municipalities with extended authority, SÚJB and NPP. This procedure serves for assuring the coordinated protective actions on the territory falling under the administration of the individual municipalities with extended authority.

Fire Rescue Services (FRS)

- FRS assures, based on the instruction from the NPP, warning of inhabitants in the EPZ by the sirens through national integrated system of warning and moreover it assures broadcasting of in-advance prepared relevant information in the Czech television and Czech radio. FRS of the region also assures for ČEZ, a. s. notifing the concerned municipalities with extended powers through its regional OIC (in accordance with the Decree No. 318/2002 Coll. as amended). For the needs of strengthening LFRS) of NPP operated by ČEZ, a, s, the alarm plans are elaborated which are a part of the off-site emergency plans of NPP. On this base also other professional units of FRS CR would be able to provide an effective aid with arrival to NPP sites in the range of 10÷60 minutes depending on their dislocation.

Integrated Rescue System

- IRS is built for the purpose of the coordinated management and coping with extraordinary situations, without further closer specification if it is an industrial accident, flood, earthquake or other natural disaster. From legislative point of view, it is covered first of all in the Act 239/2000 Coll., of the integrated rescue system and change of some Acts, as amended and Act No. 240/2000 Coll., of crisis management and change of some Acts as amended. Within IRS, the Central Alarm Plan of IRS is elaborated which may be applied if it is necessary as a consequence of EE or crisis situation or safety action and if the conditions determined by the law for the central co-ordination of rescue and liquidation work are complied with, or if the region governor, mayor of the municipality with extended authority, director of FRS of the region or commander of intervention asks through OIC FRS of the region for additional aid and for forces and means.

Ministry of Interior – General Directorate of FRS CR

- It calls and deploys through its OIC forces and means under the central co-ordination of rescue and liquidation work.

Czech Hydrometeorological Institute

- The Czech Hydrometeorological Institute assures for NPP evaluating the topical meteorological situation and processing the prognosis of further development. It transmits the basic meteorological data necessary for assessing the potential or real spreading of the radioactive leakage in NPP surroundings to respective information network of NPPs.

CR Police

- The Police cooperate in informing the inhabitants in EPZ, organization of evacuation, transport situation, guarding buildings, etc.

Medical Rescue Service (Traumatological plan)

- Based on a contract, medical service providing first aid has been established in NPPs premises with continuous emergency service which is responsible for providing medical service.

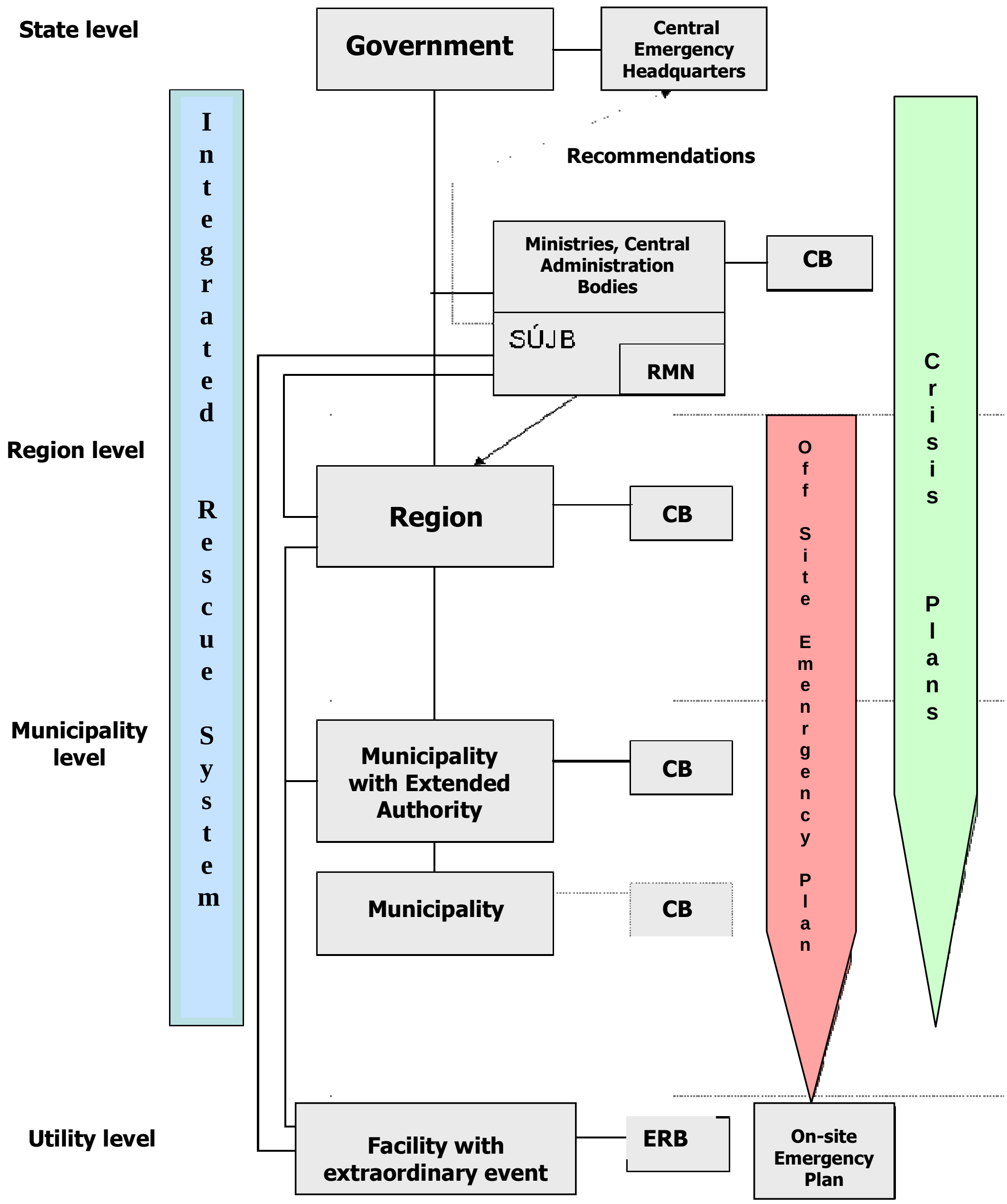

Fig. 3: Assuring off-site emergency preparedness of NPP in CR

5.2 DESCRIPTION OF ACTIVITIES PERFORMED BY THE LICENSEE

5.2.1 Survey of performed and planned activities

To increase the effectiveness of the system of coping with emergencies, the measures of emergency preparedness will be specified with more details in the following fields:

1. Assuring further alternative communication means for the communication between the intervening persons and external bodies including assuring the alternative power supply.
2. Creation of further spare ways for notification and warning and extension of the time of their back-up power supply.

3. Verification of the possibility of higher resistance and proper function of shelters for the extreme inundation and seismicity.
4. Assuring enough back-up staff for coping with the occurred EE.
5. Ability of functioning of OER outside ECC in the sites of both NPPs, additional equipment of the back-up ECC (outside the site) by appliances and other communication means.
6. Co-operation with other external components of emergency preparedness.
7. Adding additional qualified staff of OER including a more intensive OER staff training especially for the case of severe accident treatment.
8. Creation of NPP renewal team (controls activities after the extraordinary event which can cause long-term loss of production or with a risk of the complete loss of the source).

5.2.2 Further activities of the Licensee

Proposed measures	**Description of the way of implementation including possible options**
Assuring further back-up communication means for the communication between intervening persons and external bodies including assuring alternative power supply	Assuring the communication between the intervening persons and external bodies: a) inside NPP (especially control room – TSC – intervening staff) - assuring alternative power supply of telephone switcchboard located in the selected shelters; using the same sources of power supply also for emergency charging of transportable lights and internal mobile phones (solution by the installation of 2 re-charging places in NPP). - determining the concept for the communication in case of disintegration of communication network due to seismic event b) between NPP and external bodies and organizations - treatment as in the item a) with verifying the possibilities of communication of key workplaces (especially NPP – bodies of state administration) by satellite phones.
Creation of further back-up ways of notification and warning and extension of the time of their back-up power supply	In cooperation with bodies of the state administration and IRS, defining a reserve organizational solution (e.g. determining the control infrastructure for its start) in case of non-functionality of the radio and sirens in consequence of the extreme natural phenomena (equipment by mobile means - mechanical sirens, pneumatic sirens, megaphones on vehicles), incl. the way of usage and incorporating into the concerned EP documentation.
Verification of the possibility of higher resistance and proper function of shelters for case of extreme inundation and seismicity	Performing the analysis of endangering the shelters during seismicity and inundations – especially the shelter for the activity of ECB; the following implementation of measures following from the analysis

Proposed measures	**Description of the way of implementation including possible options**
Assuring the sufficient amount of the back-up staff for coping with the occurred EE	Performance: a) measures for exchange of shift personnel under aggravated availability of site, b) analyses of possibilities of usage of additional shift personnel in case of occurrence of accident at all the four units (enough persons for strategy implementation, their sheltering) c) analyses of conditions and possibilities/sufficient amount of staff for implementation of interventions as per EDMG (extensive damage mitigation guidelines) d) accelerated evacuation of persons from NPP (who do not participate in accident liquidation) in case the shelters are not capable to operate
Ability of functioning OER outside ECC in sites of both NPPs, completion of equipment of back-up ECC (outside the site) by the devices and further communication means	Verification of the possibility of activation of TSC and ERB outside ECC situated in NPP (including the possibilities of transmitting the information and assuring the necessary communication) and establishment of back- up ECCs outside both EPZ; involving the results of this verification into the EP documentation
Co-operation with other external parts of emergency preparedness	Completing agreements with external parts (e.g. IRS components with which no agreements have been concluded so far, CR Army), further institutions and nearby NPPs of aid and support of the affected Dukovany NPP or Temelín NPP.
Amending the qualified staffing of OER including a more intensive training of OER staff, especially for coping with severe accidents	Elaboration of criteria for staffing OER (employees with highest professional competence) – verification of staffing of SOER at both NPP. Re-evaluation of the concept of the staff training in the field of severe accidents and defining updated concept.
Creation of NPP Renewal Team, (it controls activities after extraordinary events which can cause a long-term loss of production or the danger of the risk of the complete loss of the source is imminent	In the scope of preparedness for the management of the post-accident conditions at NPP – setting up a „Renewal Team“.

5.2.3 Conclusions by the Licensee

Field/measure	Preliminary results
Assuring further back-up communication means for the communication between intervening persons and external bodies including assuring the alternative power supply	The negotiations with representatives of FRS CR were started about the possibilities of back-up way of communication in case of disintegration of the usually used communication network. It was agreed that for selected mobile phones priority calling in the mobile network may be set through OIC IRS for the case of EE treatment. Usage of an independent radio-station or radio-station functioning through mobile converters in co-operation with other operators of telecommunication network or with FRS CR is suggested.
Creation of additional back-up ways of notifying and extension of the time of their back-up power supply	The negotiations with representatives of FRS CR have been initiated about the possibilities of back-up ways of communication for the needs of notifying in case of disintegration of usually used communication network
Verification of the possibility of higher resistance and proper function of shelters for extreme inundation and seismicity	The analysis of endangering the shelter for various reasons is performed (already finished in the field of the extreme floods) The implementation of further back-up power supply of shelters was ordered.
Assuring sufficient amount of back-up staff for coping with occurred EE	To strengthen LFRS NPP, an alarm plan was elaborated based on which the professional units of FRS of the Czech Republic being a part of IRS would be able to provide an additional effective personal aid and material with arrival to localities within 10-60 minutes depending on the location of fire brigade unit. In the scope of IRS, among others, 6 helicopters were determined for the rescue work (Army of CR and CR Police) for the possibility of the transport of persons and load; 4 crews are in the standby mode with the possibility of activation within 10 minutes on the day and 20 minutes at night. In the shift as well as non-shift staffing additional positions will be selected for assuring the operation of units and mitigation of EE consequences which should remain at NPP (their exchange later on will be solved as well). These are e.g. employees not being in service but involved in OER.
Ability of functioning OER outside ECC at the sites of both NPP, completing equipment of a back-up ECC (outside the	The back-up emergency support centers located outside EPZ were defined. The analysis of the necessity of supplementing their equipment is performed, including the transmission of information data from the NPP and the necessary

Field/measure	Preliminary results
site) by devices/ other communication means	communication.
Co-operation with other external components of emergency preparedness.	The agreement between Temelín NPP and FRS of South-Bohemian Region on mutual assistance was revised. Moreover the revisions of similar agreements and contracts between the both NPPs and other organizations in connection with EE management on the sites of both NPPs will take place.
Supplementing the qualified staffing of OER including a more intensive training of OER staff especially for the case of managing severe accidents	The elaboration of criteria for filling individual positions of SOER is in progress as well as the revision of the conception of training of TSC staff in connection with management of severe accidents.
Creation of the team for NPP renewal (it controls activities after the extraordinary events which can cause a long-term loss of production or there is an imminent danger of the complete loss of source)	For the purpose of preparedness for management of post-accident situations at NPP, including defining criteria and effective mechanisms of deciding, a Renewal Team has been set up in ČEZ, a. s. since 2011.

5.3 DESCRIPTION OF ACTIVITIES PERFORMED BY THE STATE REGULATORY AUTHORITY

5.3.1 Survey of implemented and planned activities

State regulatory authority, i.e. SÚJB has performed:

- analysis of the current on-site emergency plan of NPP,
- inspection focused especially on the activities of the Licensee described in its respective internal documentation and concerning monitoring the radiation situation with accent on emergency monitoring.

SÚJB is planning

- to initiate revision of the safety analysis report for the purpose of updating the source terms,
- to initiate revision of the interventions levels stated in the respective internal documentation of the Licensee, serving for the classification of EE and the start intervention actions,
- start discussion on the contents of the off-site emergency plans of both EPZ,
- during the preparation of the new Atomic Act, incorporating the lessons learned in the necessary scope.

5.3.2 Further activities of the state regulatory authority

Field	Description of the way of implementation including possible options
Analysis of the current internal emergency plan of NPP	SÚJB has performed in 2011 a detailed analysis of the current on-site emergency plans of both NPPs with focus especially on the intervention procedures and deciding schemes stated there. During this analysis SÚJB came to the conclusion that a) Procedures shall be updated and specified to determine clearly the procedure for each position in OER system described in this plan (precondition - performance of the respective revision of the on-site emergency plan: 2012), b) Schemes shall be discussed with the operator and updated based on the conclusions of the discussion (start of discussion: 3-4/2012, updating – see item a))
Inspection focused especially on activities of the Licensee described in its respective internal documentation and concerning monitoring the radiation situation with accent on emergency monitoring	SÚJB performed in 2011 the inspection and among its conclusions, there is e.g. the need to perform the revision of the intervention instructions in such a way that the general intervention procedures of the activities for the employees executing the given positions within OER will be stated in the on-site emergency plan of NPP and the intervention instructions will contain the description of the sequence of partial acts with clear specification of liabilities while exchanging shifts. The intervention procedures will be revised in the scope of the total revision of the On-site Emergency Plan of NPP (see the field 1), revision of intervention instructions – estimated start 2012, estimated end 2013.
Revision of the safety analysis report for the purpose of updating source terms	In the year 2012, SÚJB will perform the revision of information on the source terms stated in the safety analysis reports of both NPPs and in view of its conclusions it will ask the NPP operator for their updating or possible amendment. Estimated updating and possible amendment of the source terms: 2013
Revision of intervention levels stated in the respective internal documentation of the Licensee, serving for the classification of EE and for starting the intervention activities	SÚJB will start in the 1st half of 2012 the control of intervention instructions for the purpose of a detailed revision of all the levels of interventions stated in the instructions. Based on the conclusions of this inspection, their specification or modification will be initiated. The supposed time of specification or modification of intervention levels set in the intervention instructions of the operator: 2013
Discussion to the contents of the off-site emergency plans of both EPZ	In 2012, SÚJB will be participating in negotiations for the revision of the contents of the off-site emergency plan of EPZ Temelín NPP, started in 2/2012 by the author of this plan, i.e. Regional Directorate of FRS in the South-Bohemian Region. SÚJB will propose in the 1st half of 2012 to the author of the off-

Field	Description of the way of implementation including possible options
	site emergency plan for EPZ Dukovany NPP to start of the discussion of the revision of its contents.
During the preparation of the new Atomic Act, incorporating the lessons learned in the necessary scope.	During the preparation of the wording of the new Atomic Act initiated by SÚJB in 2011, the acquired knowledge and lessons learned concerning the EP will be incorporated, i.e. parts of coping with radiation accidents and monitoring of radiation situation.

5.3.3 Conclusions by the state regulatory authority

Field	Description of the way of implementation including possible options
Analysis of the current on-site emergency plan of NPP	In 2011, SÚJB made a detailed analysis of the current on-site emergency plan of NPP. In 2012 SÚJB plans that the Licensee a) will update and specify the intervention procedures in such a way that the procedure for each position in OER system described in this plan is clearly defined (expected revision of the on-site emergency plan: 2012), b) in connection to conclusions of the discussion with SÚJB the Licensee will modify or update schemes of deciding contained in the plan (start of discussion: 3-4/2012, expected updating – see above (a)
Inspection focused especially on activities of the Licensee described in its respective internal documentation and concerning monitoring the radiation situation with accent on the emergency monitoring	SÚJB performed in 2011 the inspection and among its results the need of revision of intervention instruction was identified, in such a way that the general intervention procedures of activities for employees executing the given position within OER will be stated in the on-site emergency plan of NPP, and the intervention instructions shall involve the description of the sequence of the partial actions with clear specification of liabilities in case of shifts exchange. The intervention procedures will be revised in the scope of the total revision of the on-site emergency plan of NPP (see the field 1), revision of intervention instructions – supposed start 2012, expected end 2013.
Revision of safety analysis reports for the purpose of updating source terms	There are no results (even not preliminary results) so far.
Revision of intervention levels contained in the respective internal documentation of the	There are no results (even not preliminary results) so far.

Field	**Description of the way of implementation including possible options**
Licensee, serving for the classification of EE and for starting the intervention activities	
Discussion to the contents of the off-site emergency plans of both EPZ	There are no results (even not preliminary results) so far.
During the preparation of the wording of the new Atomic Act – incorporation of acquired knowledge and lessons learned in the necessary scope	In 2011, the base documents were prepared for the 1st version of the draft of the wording of the new Atomic Act. In 2012, the internal (at SÚJB) commenting of the 1st draft and elaboration of the following version are planned.

5.4 FINAL SUMMARISATION OF THE CHAPTER 5

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 5.2.1) Activity - Taken? - Ongoing? - Planned?	(Item 5.2.2) Schedule Or Milestones for Planned Activities	(Item 5.2.3) Results Available - Yes? - No?	(Item 5.3.1) Activity - Taken? - Ongoing? - Planned?	(Item 5.3.2) Schedule Or Milestones for Planned Activities	(Item 5.3.3) Conclusion Available - Yes? - No?
Topic 5 Emergency Preparedness and Emergency Response						
Assuring other alternative communication means for the communication among intervening persons and external bodies including assuring the alternative power supply	Ongoing	2015	No	Planned	Inspection 2016	No
Creating the concept of additional alternative ways of informing and warning	Ongoing	Not determined so far	No	Planned	Inspection 2013	No
Additional alternative ways of notifying and warning and extension of the time of their back-up power supply	Ongoing	Stage 1 2014	No	Planned	Inspection 2015	No
Verification of the possibility of higher resistance and proper function of shelters for the extreme inundation and seismicity – analysis of shelter endangering	Ongoing	Not determined so far	No	Planned	Inspection 2013	No
Verification of the possibility of higher resistance and proper function of shelters for the extreme inundation	Ongoing	2015	No	Planned	Inspection 2016	No

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 5.2.1) Activity - Taken? - Ongoing? - Planned?	(Item 5.2.2) Schedule Or Milestones for Planned Activities	(Item 5.2.3) Results Available - Yes? - No?	(Item 5.3.1) Activity - Taken? - Ongoing? - Planned?	(Item 5.3.2) Schedule Or Milestones for Planned Activities	(Item 5.3.3) Conclusion Available - Yes? - No?
and seismicity - implementation of the back-up power supply						
Assuring sufficient number of back-up staff for managing EE	Ongoing	Not determined so far	No	Planned	Inspection 2013	No
Ability of functioning OER outside ECC in the sites of both NPPs, completion of the equipment of the back-up ECC (outside the site) by the devices and other communication means	Ongoing	Not determined so far	No	Planned	Inspection 2013	No
Co-operation with other external bodies of emergency preparedness.	Ongoing	2012	No	Planned	Inspection 2013	No
Adding qualified experts to OER including a more intensive training of OER staff, especially for management of severe accidents	Ongoing	2013	No	Planned	Inspection 2014	No
Creating the team for NPP renewal, (it controls activities after the extraordinary events which may cause a long-term loss of production or the risk of complete loss of source)	Adopted	2011	Yes	Planned	Inspection 2012	No
Analysis of the				Performed	2011	Yes

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 5.2.1) Activity - Taken? - Ongoing? - Planned?	(Item 5.2.2) Schedule Or Milestones for Planned Activities	(Item 5.2.3) Results Available - Yes? - No?	(Item 5.3.1) Activity - Taken? - Ongoing? - Planned?	(Item 5.3.2) Schedule Or Milestones for Planned Activities	(Item 5.3.3) Conclusion Available - Yes? - No?
current on-site emergency plan of NPP						
Analysis of the current on-site emergency plan of NPP – incorporating the findings of the analysis	Ongoing	2012	No			
Inspection focused on the activities of the Licensee described in its respective internal documentation concerning monitoring the radiation situation with accent on emergency monitoring				Taken	2011	Yes
Inspection focused on the activities of the Licensee described in its respective internal documentation concerning monitoring the radiation situation with accent on emergency monitoring – incorporating findings from the performed inspection	Ongoing	2012 - 2013	No			
Revision of safety analysis report for the purpose of updating the source terms				Planned	2012	No

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 5.2.1) **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 5.2.2)** **Schedule Or Milestones for Planned Activities**	**(Item 5.2.3)** **Results Available** **- Yes?** **- No?**	**(Item 5.3.1)** **Activity** **- Taken?** **- Ongoing?** **- Planned?**	**(Item 5.3.2)** **Schedule Or Milestones for Planned Activities**	**(Item 5.3.3)** **Conclusion Available** **- Yes?** **- No?**
Revision of safety report for the purpose of updating the source terms – incorporating results	Planned	2013	No			
Revision of intervention levels contained in the respective internal documentation of the Licensee, serving for the classification of the EE and start of intervention activities				Planned	2012	No
Revision of intervention levels contained in the respective internal documentation of the Licensee, serving for the classification of the EE and start of intervention activities – updating respective documentation	Planned	2013	No			
Discussion to the contents of the off-site emergency plans of both EPZs				Planned	2012	No
In the scope of the preparation of the new Atomic Act, incorporation of acquired				Started	2011	No

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 5.2.1) Activity - Taken? - Ongoing? - Planned?	(Item 5.2.2) Schedule Or Milestones for Planned Activities	(Item 5.2.3) Results Available - Yes? - No?	(Item 5.3.1) Activity - Taken? - Ongoing? - Planned?	(Item 5.3.2) Schedule Or Milestones for Planned Activities	(Item 5.3.3) Conclusion Available - Yes? - No?
knowledge and lessons learned						

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6. INTERNATIONAL COOPERATION

6.1 INTRODUCTION

Besides the individual activities of the international co-operation analyzed hereinafter in the part of the Licensee as well as state regulator, it is necessary to mention at the beginning the process initiated directly by the accident of Fukushima Daiichi NPP in Japan.

This event initiated in the European Union the requirement on the assessment and evaluation of resistance of the European NPPs towards extreme and very improbable phenomena for which the NPP design may be not sufficiently prepared as their occurrence was not supposed during the design of these facilities.

The requirement of the European Commission (EK) on the performance of the „stress tests“ was sent to member countries of EU on May 24, 2011. The objective of these tests was to identify existing safety margins and to set the time during which the emergency situation develops into the severe accident with the subsequent degradation of fuel with a large ¨release of radioactivity into the surroundings. The technical content of the stress test was defined by the association of the European Nuclear Safety Regulators Group – ENSREG. The requirement was elaborated in detail by ENSREG group in the form of the recommendation of a detailed structure of evaluating reports of nuclear power plant operators and national reports elaborated and submitted by the national regulatory bodies.

The stress tests are part of the comprehensive safety evaluation of NPP as follow-up of the international documents devoted to the given event (e.g.: WANO SOER 2011-2, Fukushima Daiichi Nuclear Station Fuel Damage Caused by Earthquake and Tsunami, March 2011; WANO SOER 2011-3, Fukushima Daiichi Nuclear Station Spent Fuel Pool/Pond Loss of Cooling and Makeup, August 2011; INPO Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Station, November 2011; IAEA International fact finding expert mission of the Fukushima-Daichi NPP Accident Following the Great East Japan Earthquake and Tsunami, June 16, 2011; US NRC Recommendation for enhancing reactor safety in the 21^{st} century, July 12, 2011).

The results and conclusions following from these „stress tests“ are analyzed in detail in the respective chapters of this Extraordinary National Report of CR – especially in chapters 1, 2 and 3.

6.2 DESCRIPTION OF ACTIVITIES PERFORMED BY THE LICENSEE

6.2.1 Survey of the implemented and planned activities

The Licensee for the operation of the nuclear installation - ČEZ, a. s., is active member of a number of professional international organizations and association in nuclear power and is actively involved in many international programs and exchange of operational experience. It uses practically all the available international sources for maintaining a high state of knowledge and is involved in the programs for enhancement nuclear safety level, know-how, knowledge of state-of-the-art technology and usage of good practice from all over the world.

It uses the membership of CR in international organizations as the IAEA, OECD and it is a member of a number of associations of operators (e.g. WANO, EUR, ENISS, NUMEX etc.).

Utilization of experience of other NPP operators (feedback of external operational experience), which is a part of the wider WANO Operating Experience program is an important source of incentives for positive changes increasing the operational safety. ČEZ, as the active member of the international associations in nuclear power not only utilizes but also contributes by its experience into the international databank of best practices; by participation of its experts in international missions, seminars, workshops and in technical negotiations it increases the world-wide operational and safety know-how.

„International co-operation“ is, from the viewpoint of organization and management of the Production division of ČEZ company defined as independent process. The objective of this process is creation of effective contacts abroad, acquiring and handing over information and know-how, exchange of experience / information with foreign parties with the aim of implementing the safety policy and quality assurance policy of ČEZ, a. s., and practical increase of safety, reliability and effectiveness of the operation its NPPs. The objectives of co-operation with international organizations are as follows:

- To be well informed about the current developments in the nuclear power, to assess existing risks and experience of other operators, to utilize the opportunities for improving the nuclear power plant safety.
- To co-operate actively in the selected professional international associations / organizations for assuring technical support and the world know-how
- To acquire and to implement know-how, good practices, best experience and the results of the operational experience feedback, benchmarking, new procedures and technologies at NPPs of ČEZ, a. s., (to implement the system of „Learning organization“).
- To prevent the isolation of nuclear power plants operation in the CR from the developing nuclear community in the world (to keep pace with the world trends of branch development).
- To present and to share operating experience with the aim of increasing the safety level of the whole branch (collective liability for the high level of nuclear safety also on other nuclear facilities in the world).
- To build and to maintain long-term professional and well as personal relationships with the foreign power engineering companies, similar (as for design) nuclear power plants, organizations and institutions.
- To educate the managers and experts in communication with foreign countries, with aim of developing professional knowledge and international information exchange.

ČEZ is member of the following professional associations:

- WANO (World Association of Nuclear Operators),
- FORATOM (European Atomic Forum),
- ENISS (Association of operators for harmonizing European NS standards),
- ENS + ČNS (European and Czech Nuclear Society),
- NucNet (The Communications Network for Nuclear Energy and Ionizing Radiation),
- ENC, European Nuclear Council (Association of CEO nuclear operators in Europe),
- NUMEX (Nuclear operators platform for maintenance of nuclear installations),
- Eurelectric (Union of Electricity Industry),
- WNA (World Nuclear Association),
- EPRI (Electric Power Research Institute),
- VGB (German scientific-technical organization, a.g. for nuclear power engineering),
- Chamber for economic contacts and SNS,

- EUR (Association of West-European operators with the aim of standardization of safety requirements for nuclear reactors of new generation).

6.2.2 Further steps of the Licensee

6.2.2.1 International sharing of operating experience

Nuclear power plants in the Czech Republic (Nuclear Power Plant Dukovany - EDU and Nuclear Power Plant Temelín - ETE) are involved in the international system of sharing operational experience (IAEA, WANO). In parallel with this they are orientated on and have direct contacts first of all with identical types of nuclear power plants in Slovakia, Hungary, Finland, Ukraine and Russia. Besides this, the ČEZ, a.s. experts participate in the work of working groups of other professional organizations, such as e.g. EUR, ENISS, ENC, FORATOM, Eurelectric, WNA, etc.

The main task of this co-operation is the transfer and utilization of operational experience and technical information of nuclear power plant operators in practice of both nuclear power plants. Selected important information about events at other nuclear facilities and international experience from sources like WANO, IAEA, event. INPO,… are monitored and included into the program of the Event Committee, meetings of the Production division director, meetings of EDU/ETE management (and subsequently departments), Committee for Safety of the Production division and Committee for Safety of EDU (ETE). The most important experience with a possible impact on the operation or safety of the nuclear power plants is implemented in the form of corrective measures. They focus first of all on the training of personnel, maintenance and improving inspection activities. All the acquired information on external events is stored in the database supported by special software and they are utilized by the specialists of individual departments as technical support in dealing with problems. The staff concerned is directly informed of the most important events on the foreign power plants at the training days. The tasks and measures resulting from these events are implemented and their effectiveness is evaluated. Safety events and operational experience from non-nuclear operations of ČEZ, a. s., are transmitted at EDU and ETE in the standard working way of communication inside ČEZ, a. s.

Vice versa, operational experience of EDU/ETE is handed over to other NPP operators either by the direct contact or by elaborating 4 to 6 detailed reports annually on the most important events with the analysis of their root causes. These reports are then included into the international WANO network or they are submitted to SÚJB for distribution via Incident Reporting System (IRS) network of the IAEA.

6.2.2.2 Multilateral systems

- Operating Experience Program (WANO) **-** Program of utilization of the external feedback / informing about the events: reports on events at NPPs from the all over the world and/of from ČEZ. Usually, one report on the event/unit/year, corrective measures for all recommendations issued by WANO.
- Direct information exchange of (WANO) – operational safety indicators of WANO, mission of technical support for the selected topics, technical benchmarking with other NPP operators in the world, restricted communication inside WANO network (several dozens discussion columns for dealing with technical issues, questions and answers).
- INES/IRS – elaborating and submitting reports for SÚJB and then into international databases INES and IRS.

6.2.2.3 International Peer Review

The resources for enhancing safety include the outputs of the evaluation program the IAEA, OECD-NEA and WANO. An example may be a comprehensive program of the IAEA devoted to assessment of safety of the "Russian" reactor types, including the VVER 1000 (IAEA-EBP-WWER-05), which resulted in a number of safety issues. Furthermore, among such resources one can consider also outcomes of independent international audits, in particular missions OSART and SALTO of the (IAEA) or WANO Peer Review, which are processed into the form of action plans containing the proposed corrective measures and new WANO tasks.

WANO

ČEZ, a. s., as an active member of WANO invites regularly the international program of partner inspections (WANO Peer Review - WPR). These inspections are performed by the international expert teams from various professional organizations and nuclear power plants operated in other countries and cover 10 standard areas (Effectiveness of Organization and Management, Operation, Maintenance, Engineering Support, Utilization of Operational Experience, Radiation Protection, Chemistry, Personnel Training and Qualification, Fire Safety, Emergency Preparedness). Program of partner inspections supposes one WPR in 4 years at each NPP and after two years a subsequent or other independent inspection.

ČEZ' employees also participate in this program in international teams visiting other NPPs. Such participation contributes to the transfer of know-how, safety benchmarking and enhancing of safety level in the world.

The first WANO Peer Review at Dukovany NPP took place in 1997, the second WPR mission in 2007, with the follow up mission in January 2009. The missions confirmed a high level of safety of operation of Dukovany NPP. The follow up inspections confirmed that all recommendations for improvement were implemented or were in the high stage of completion.

In November 2011, Temelín NPP hosted already the third WPR mission. Preceding missions took place in the years 2004 and 2006. WPR appreciated in Temelín NPP the high professionalism of the staff and the achieved safety level of the power plant. The mission summarized its conclusions in the form of 17 minor recommendations for improvement and also highlighted 3 examples of good practice for other NPP operators around the world.

The next WANO Peer Reviews in Dukovany NPP and Temelín NPP (follow-up mission) are planned for the years 2012-2013.

IAEA

The Czech Republic invites regularly international missions of the IAEA. In case of NPP these are following types of missions: OSART, ASSET, SALTO, IPERS, Safety Issues, IPPAS, Site SR Design, LBB assessment, Fire Safety, PSA and Seismic SBSA. Dukovany NPP has hosted in total 15 and Temelín NPP 21 international missions.

The first international OSART mission was organized in Dukovany NPP in 1989 and the last one took place in June 2011 (in between there were missions in the years 1991, 2001, 2003, including follow-up missions). In June 2011 ČEZ invited to Dukovany NPP the last OSART mission of the standard scope, i.e. it covered all the following fields:

- Management, organization and administration (3 experts),
- Training and qualification (1 expert),
- Operation (2 experts),

- Maintenance (1 expert),
- Technical support (1 expert),
- Feedback of operational experience (1 expert),
- Radiation protection (1 expert),
- Chemistry (1 expert),
- Emergency preparedness (1 expert).

The results of the mission were very positive and EDU was evaluated as NPP with very high level of safety. This was reflected also in the number of recommendations (3), proposals (11) and internationally recognized good practices (10).

The first OSART mission at Temelín NPP was organized in 1990. It was the pre-commissioning mission and the follow-up took place in 1992. The first mission on the full scope took place in 2001 and the follow-up in 2003. During the follow-up mission the most recommendations and proposals from previous mission was classified as “fulfilled” or in “satisfactory progress of solution” and the OSART team appreciated a considerable improvement of operational safety, state of implementation of recommendations and the overall increase of the power plant efficiency. The next OSART mission is planned for Temelín NPP in 2012.

6.2.2.4 International recommendations and technical standards

An important source of the recommendations are publication of the IAEA, in particular Safety Series: TECDOCs, Safety Fundamentals, Safety Requirements, Safety Guides, INSAG reports including information data bases, e.g. AIRS - Advanced Incident Reporting system database, OSMIR – OSART Mission Results databases, etc.

Another external source of information is WANO, providing a number of products, contributing to the increase of the safety and effectiveness of NPP operation: Guidelines, Performance Objectives and Criteria, Just-in-Time, lessons learned from events (Significant Event Reports/Significant Operating Experience Reports), methodology of Self-Assessment, Excellence in Human Performance, Operating Decision Making (ODM), Hot Topics, etc.

6.2.2.5 Active participation in international meetings

The employees of the Licensee participate actively in the international professional events as e.g. WANO technical support missions, WANO workshops and seminars, technical negotiations which represent a valuable source of external experience:

WANO - Program of technical qualification development – it includes seminars, technical meetings and workshops (the organization of min. 1-2 seminars annually in the CR, and the participation in ca 2÷3 seminars abroad is supposed) – participation of ČEZ experts depending on qualification and competence.

WANO technical support missions–a short missions with the participation of experts from other NPPs; they help to suggest corrective measures or solutions.

Examples of technical support missions at EDU and ETE in the last years:

- Taking Operational Decisions (in critical situations Operational Decision Making ODM) – (Mr. Tim Martin, WANO Atlanta Centre, training June 2005). Subsequently EDU elaborated and implemented relevant operating instruction.
- Improving the Human Factor (Mr. Tim Martin, WANO Atlanta Centre + team from the USA and Canada, June 2008). ČEZ implemented the methodology of improving the quality of human performance (QLV).

- WANO-Conference of NPP managers, Nov. 10-12, 2008, Prague, Czech Republic – Exchange of experience among NPP directors from European countries.
- Implementation of Methodology of Self-Assessment (Mr. Steve Milton, British Energy, July 2009). Methodology of self-assessment has been implemented in Dukovany NPP since January 1, 2010.
- Self-Assessment and Program of Remedial Measures (Self-Assessment and Corrective Action Programs), NPP Temelin (November 9. – 13, 2009).
- International seminar for middle management of nuclear power plants: safety culture, human performance quality, instruments for preventing failures, guiding people to safety (6th Leadership Workshop for Middle Managers), WANO Paris Centre, April 6.-9, 2010, Prague, Czech Republic – participation of EDU as well as ETE managers.
- Implementation of Near-Miss program (Mr.Conrad Dubé, WANO Paris Centre+ WANO team, October 2010), Dukovany NPP.
- Opportunities for Increase of Load Factor by Higher Effectiveness of Outages, Technical support mission WANO, March 22-25, 2011, Temelín NPP.
- IAEA (IAEA) – Technical Meeting on Evaluation of Effectiveness of Operational Safety Review Services and their Future Evolution, 1–4 November 2011, Vienna, Chairman: Koen van Beveren. as reaction on the accident at NPP Fukushima Daiichi; number of negotiations took place with the operators of other NPPs type VVER - aimed to harmonize practices in individual countries and to draw lessons mutually from best practices
- VVER 440-V213 club with participation of partner NPP Jaslovské Bohunice, Mochovce, Paks and Loviisa, was renewed in 2011.
- Meeting of management of partner's NPP on V-213 modernization, requirements of WENRA and regulatory requirements, January 20- 21, 2011, Prague.
- Initiation of EU stress tests, May 12, 2011, ČEZ, Prague.
- Strategic spare parts for VVER 440, August 24, 2011, NPP Paks, Hungary.
- Presentation of achieved results and harmonization of reports to EU stress tests. October 12, 2011, Paks, Hungary.

6.2.3 Preliminary results of Licensee's activities

To assure a high safety level, ČEZ, a. s., issued its „Safety and Environment Protection Policy“ (ČEZ_PRGR_1008), which contains in ten chapters the key principles for achieving defined goals in the given area. Especially the chapters 3, 4, 6 and 7 are devoted to the transfer of knowledge and utilization of experience from power plants for assuring the sufficient safety level.

6.3 DESCRIPTION OF ACTIVITIES PERFORMED BY THE STATE REGULATORY AUTHORITY

6.3.1 Survey of the implemented and planned activities

Bilateral co-operation on various levels represents another important international activity. It is organized by SÚJB as well as other governmental bodies (Ministry of Industry and Trade, Ministry of Education, Youth and Sports, Universities, etc.) as well as by both NPPs in co-operation with IAEA and other international organization.

Preparation and signing of international agreements also belongs to important state administration activities. In view of the given procedure and the fact that most of these activities in nuclear area are coordinated by the state regulatory authority (the rest by other bodies of the state administration), they are summarized in an independent sub-chapter.

6.3.2 Future activities of the state regulatory authority

6.3.2.1 Contractual base of international co-operation

International agreements and conventions are the legal base of the international co-operation and may be split into four groups: agreements connected with EU including main legal acts of the acquis communautaire directly influencing the regulatory framework, general treaties of United Nations, bilateral international agreements and agreements between the regulators.

Agreements and further legal acts connected with EU

Treaty establishing the European Atomic Energy Community

Council Directive 2011/70/ EURATOM of July 9, 2011, establishing a Community Framework for the Responsible and Safe Management of Spent Fuel and Radioactive Waste,

Council Directive 2009/71/ EURATOM of June 25, 2009, establishing a Community Framework for the Nuclear Safety of Nuclear Installations,

Council Directive 96/29/EURATOM of May 13, 1996, laying down Basic Safety Standards for Protection of Health of Workers and General Public against Danger Arising From Ionising Radiation,

Council Directive 89/618/EURATOM of November 27, 1989 on Informing General Public about Health Protection Measures to be Applied and Steps to be Taken in Event of Radiological Emergency**,**

Council Directive 2006/117/ EURATOM of November 20, 2006 on Supervision and Control of Shipments of Radioactive Waste and Spent Fuel.

International Conventions

CR is a long-term contracting party to the following international conventions:

- The Convention on the Physical Protection of Nuclear Materials,
- The Convention on Early Notification of a Nuclear Accident,
- The Convention on Assistance in the Case of a Nuclear or Radiation Emergency,
- Convention on Nuclear Safety,
- The Comprehensive Nuclear Test Ban Treaty,
- Joint Convention on the Safety of Spent Fuel Management and on the Safety, of Radiological Waste Management,
- The Treaty on the Non-Proliferation of Nuclear Weapons (NPT),
- The Convention on Environmental Impact Assessment in a Transboundary Context,
- Vienna Convention on Civil Liability for Nuclear Damage,
- Convention on Supplementary Compensation for Nuclear Damage.

Bilateral agreements on co-operation in nuclear safety

CR has concluded international agreements (on governmental level) on co-operation, or exchange of information including emergency information in nuclear safety area with the following countries: Australia, Bulgaria, India, Canada, Korea, Hungary, Germany, Poland, Austria, Russian Federation, Slovakia, Ukraine, and USA.

Bilateral arrangements between regulators

CR has concluded international agreements (on the level of regulatory authorities) on co-operation or exchange of information including crisis information in nuclear safety with the regulatory bodies of the following countries: Finland, France, Canada, Korea, Hungary, German, Rumania, Russian Federation, Slovakia, Spain, Great Britain, Ukraine, USA. They are rather of "working" character, not enforceable as per the international law. In some cases, these are arrangements focusing on a very narrow area.

6.3.2.2 Bilateral co-operation

The most of bilateral agreements and arrangements provide a legal base for the co-operation, which has an occasional character. Regular consultations about the safety of the nuclear installation take place with the following states: Hungary, Germany, Poland, Austria, Slovakia, and Slovenia. The programme of the meetings includes:

- changes in the regulatory organization of and legislation,
- safety analysis of events and the regulatory position to taken measures,
- modernization of NPP, new nuclear units and their licensing,
- spent fuel and nuclear waste management,
- monitoring of radiation and emergency preparedness.

These topics were discussed in 2011 in the light of accident at NPP Fukushima Daiichi.

6.3.2.3 Multilateral international co-operation and international working groups

CR participates in the international co-operation in the frame of the following international and multinational organizations, institutions and associations: EU, IAEA, OECD, WENRA, WWER Forum, and NEWS.

CR utilized the current relations and membership in the international institutions for sharing the preliminary conclusions and lessons learned from the events at NPP Fukushima Daiichi and for an active participation in common new working groups focused specially on this issue. CR accepted a principal obligation on performing stress tests at Czech NPPs according international (EU) rules, elaboration and presentation of the national report and participation in the international evaluation process (Stress Tests Peer Review).

CR utilizes the gained internationally experience in the extensive revision of legislation in 2012 and preparation for the international IRRS mission which should take place in 2013.

Co-operation in the Scope of EU

CR (SÚJB, Ministry of Industry and Trade and other bodies of the state administration) co-operates with other member states in the frame of EU Council (Atomic Question Group - AQG) as well as with working and consulting groups set up by EC (ENSREG, INSC Committee).

The working group for the field of nuclear safety (AQG) is the working group of EU Council, dealing especially with topics associated with peaceful use of nuclear energy and radiation protection; on a working level; legislation under preparation by EURATOM it also discusses there.

The working group of European Nuclear Safety Regulators Group, ENSREG is independent group representing EU regulatory authorities in the field of nuclear safety on the highest level, created by the decision of the European Commission and aiming at achieving of a common consensus on various matters. CR participates in the work in all the three working groups, in particular:

- Nuclear safety,
- Safe management of spent fuel and nuclear waste,
- Transparency and public relation.

Instrument for Nuclear Safety Cooperation (INSC) Committee is a working group of EC for the co-operation of EC and member states focused on preparation and approving projects for assistance to third countries in the nuclear safety. INSC is a follow-up of previous TACIS program.

SÚJB experts participate also in implementing INSC projects for support to regulatory bodies in the third countries either operating or planning the use of nuclear reactors for the electricity production. SÚJB participates in the projects in Armenia, Egypt, Jordan and in the Ukraine.

Co-operation with IAEA

SÚJB's Chairperson serves at present as the vice-chairman of the Board of Governor, who is also a member of SAGTAC.

SÚJB participates in all the four standardizing commissions for creation of international standards, in particular: Nuclear Safety Steering Committee (NUSSC), Radiation Safety Steering Committee (RASSC), Safety of Radioactive Waste Steering Committee (WASSC) and Safe Transport of Radioactive Materials Steering Committee (TRANSSC).

CR provides a number of experts for the services provided by IAEA to member countries (IRRS, OSART missions, etc.) and it participates in the Technical Co-operation Program of IAEA, as recipient as well as provider (financially as well as by expertise).

CR participated actively in the „Ministerial Conference on Nuclear Safety“ reacting on the accident of NPP Fukushima Daiichi. IAEA created, in co-operation with member states, and started to implement “Action Plan on Nuclear Safety“. One of the tasks of the international community is to re-evaluate the current frame of nuclear safety and emergency preparedness as well as emergency preparedness and to strengthen it based on the lessons learned from the thorough evaluation of the causes and course of Fukushima accident. The experts from SÚJB and further relevant institutions from the CR get actively involved into all the associated activities not only in the frame of the IAEA, but also of other professional organizations.

Co-operation in the Frame of NEA/OECD

In the NEA/OECD, SÚJB representatives participate in the work of the working groups of the following Committees:

- ***Committee on Radiation Protection and Public Health (CRPPH)***

In 2011, SÚJB became involved into the activities of the Committee on Radiation Protection and Public Health (CRPPH). SÚJB's representative is a member of CRPPH Executive Board determining the topical focuses of activities of the Committee. The experts for radiation protection participated actively in the meeting of CRPPH committee and also in the activities of its working groups - Information System on Occupational Exposure for evaluation of exposures in nuclear installations and especially Expert Group on Occupational Exposure (EGOE) dealing with implementation of the new recommendations of the International Commission on Radiological Protection especially in the field of regulation of professional exposures and their optimization. Attention is devoted also to determining the requirements on radiation protection for new nuclear reactors.

- ***Committee on Nuclear Regulatory Activities (CNRA)***

The objective of CNRA is the exchange of experience from the practices of nuclear safety regulators; it has working groups on:

- inspection practices of regulators (WGIP),
- safety requirements on new nuclear sources (WGRNR) and
- operating experience (WGOE).

As reaction on the accident in the NPP Fukushima Daiichi, a special working group was established within the CNRA, dealing with aspects of this accident and aimed in providing a feedback from this event for the regulators (with participation of SÚJB representative). In co-operation with other committees of NEA/OECD a special group was created coordinating these activities of all the working parties with the participation of the SÚJB representative.

- ***Committee on Safety of Nuclear Installations (CSNI)***

The objective is to assist member states to maintain and to develop the scientific and technical knowledge base necessary for the evaluation of safety of nuclear reactors and fuel cycle facilities. The CSNI committee has a number of permanent working groups, in particular: on

- Integrity of Components and Structures (IAGE)
- Analysis and Management of Accidents (WGAMA)
- Risk Assessment (WGRISK)
- Human and Organisational Factors (WGHOF)
- Fuel Safety (WGFS)
- Fuel Cycle Safety (WGFCS).

At the meeting of WGHOF in 2011 the exchange of information took place between representatives of individual states on how the individual regulators reacted on the event at NPP Fukushima Daiichi and the Group elaborated the list of topics and issues arising from this event.

Co-operation in the Association of European Regulators WENRA

Fukushima accident seriously influenced WENRA association activity in the last year. In connection with the decision of EU Council, the Association was authorized to elaborate a technical framework and the scope of „stress tests“ for the nuclear power plants operated in EU countries. The draft was created by the group of experts including SÚJB representative and in May 2011, it was approved by the European Nuclear Safety Regulators Group (ENSREG). Evaluation of the nuclear power plant as per „Fukushima scenario“ was started in June. WENRA dealt also with the issue of organization of independent peer reviews, being the next stage of the „stress tests“ after performing the safety evaluation on the national level.

SÚJB representatives participate in the plenary meetings as well as working groups of WENRA:

In March 2011 Reactor Harmonization Working Group (RHWG) published the report on the state of harmonization of the safety of nuclear reactors operated in individual member states of WENRA. In May 2011, it published also the results of its survey from the year 2009 – survey of national practices in the field of assuring the long-term operation (LTO) of nuclear power plant. RHWG provided also the technical support for the creation of methodology of „stress tests“ and planning the following peer reviews.

Working Group on Waste and Decommissioning (WGWD) continued the harmonization of requirements on safety of management of radioactive waste and spent nuclear fuel, decommissioning of nuclear facilities and on for radioactive waste repositories.

The activity of WENRA Inspection Group (WIG), examining the possibility of harmonization of inspection activities in WENRA member states, was, after fulfilling its original mandate – submitting the report on the present practices in November 2011 – temporarily stopped in view of the big workload of all the experts participating in safety assessments after Fukushima accident. For the same reason also the activity of the working group dealing with the issue of safety of research reactors was slowed down.

Co-operation in the Forum of the State Nuclear Safety Authorities of the Countries Operating WWER Type Reactors

Annual plenary meeting of Forum is devoted mainly to operational safety and nuclear legislation and its putting into effect. Besides the usual national presentations to the state of nuclear safety and activity of regulatory bodies in 2011, the focus was on events at the nuclear power plant Fukushima Daiichi and the attitude of individual regulatory bodies to performing stress tests. In the Forum framework, there are following three working groups:

- Regulatory aspects of organizational, management and safety culture related issues of NPPs (led by Hungary),
- Probabilistic safety assessment (led by Finland) and
- Working group on Determining of requirements for quality of fabrication and justification of operation's safety of nuclear fuel (led by Russia).
- CR takes part in the work in all three groups. A working group on NPP life-time extension is considered.

6.3.2.4 International Peer Reviews

In the nineties, CR received a number of international evaluating missions (peer reviews), focusing first of all on the safety of nuclear installations. They are described in the preceding sub-chapter as operator's activity.

IRRT mission focusing on the quality of the regulatory activities took place in SÚJB in 2001. The overall positive evaluation resulted in five priority areas of further development, i.e.:

- Development of inspection plan involving self-assessment of the operator,
- Formalization of evaluation of training/drills in emergency preparedness,
- Performance of regular emergency drills of approved emergency plans,
- Strengthening and improving of capacities for evaluating safety culture and human factor and use of probability safety assessment in accordance with the international practice,
- Completion and implementation of electronic database of SÚJB decisions.

Next IRRT mission is planned for 2013, preparatory work in SÚJB has already started since 2011.

6.3.2.5 Sharing of operating experience

CR participates in all the formal international systems established for this purpose, i.e. in the system of notification and evaluation of events (INES) as well as in the system of sharing operational experience (IRS).

Moreover, it co-operates with „Clearing House" in JRC Petten.

Besides this, the operational experience and standpoints of regulators are informally shared in the frame of bilateral and multilateral contacts. In the first case, the operational experience is the main topic of regular meetings, in the second case; it is working group WGOE of the CNRA (NEA/OECD).

6.3.2.6 Applying IAEA standards

IAEA standards are normally used in the CR and their requirements are included into the existing legislation. They are explicitly incorporated also into the prepared draft of the new Atomic Act.

In relation to the accident at NPP Fukushima Daiichi the IAEA plans to perform an extensive revision of its standards. CR will participate in this activity through its specialists.

6.3.3 Conclusions by the state regulatory authority

From SÚJB viewpoint European stress tests, i.e. evaluation of self-assessment of the operator, elaboration of the national report and participation in the peer-reviews of member states of EU has been the main international activity reacting on the accident at NPP Fukushima Daiichi. There are ongoing discussions in a number of working groups on this topic and it is necessary to wait for their conclusions. The conclusions from these activities will be known in the second half of the year.

6.4 FINAL SUMMARY OF CHAPTER 6

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 6.2.1) Activity - Taken? - Ongoing? - Planned?	(Item 6.2.2) Schedule Or Milestones for Planned Activities	(Item 6.2.3) Results Available - Yes? - No?	(Item 6.2.1) Activity - Taken? - Ongoing? - Planned?	(Item 6.2.2) Schedule Or Milestones for Planned Activities	(Item 6.3.3) Conclusion Available - Yes? - No?
Topic 6 International Cooperation						
Self-assessment performed by the Licensee in the scope of stress tests	Taken	finished October 31, 2011	Yes			
Inspection of self-assessment of the Licensee performed by the SÚJB				Taken	Final report of CR issued on 31st December 2011;	Yes
Peer Review of the National report of CR on stress tests	Planned	March 26.-29, 2012	No	Planned	March 26.-29, 2012	No
Presentation of achieved results by the Licensee and harmonization of reports on stress test	Taken	October 12, 2011	Yes			
Renewal of the Club of VVER 440 operators	Taken	2011	Yes			
Discussion about the consequences of the accident at NPP Fukushima at VVER Forum				Taken	July 2011	Yes
Discussion about the consequences of the accident at NPP Fukushima in WENRA				Taken	first half of 2011	Yes
Discussion about the consequences of the accident at NPP				Taken	first half of 2011	Yes

Activity	Activities by the Licensee			Activities by the State Regulatory Authority		
	(Item 6.2.1) Activity - Taken? - Ongoing? - Planned?	(Item 6.2.2) Schedule Or Milestones for Planned Activities	(Item 6.2.3) Results Available - Yes? - No?	(Item 6.2.1) Activity - Taken? - Ongoing? - Planned?	(Item 6.2.2) Schedule Or Milestones for Planned Activities	(Item 6.3.3) Conclusion Available - Yes? - No?
Fukushima in OECD/NEA committees						
Discussion about the consequences of the accident at NPP Fukushima at Ministerial Nuclear Safety Conference				Taken	June 2011	Yes

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