REPUBLIC OF SLOVENIA
MINISTRY OF THE ENVIRONMENT AND SPATIAL PLANNING
SLOVENIAN NUCLEAR SAFETY ADMINISTRATION

SLOVENIAN NATIONAL REPORT ON NUCLEAR STRESS TESTS

Final Report

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Cover page picture:

Slovenian Nuclear Power Plant Krško in 1990 during the worst flood in its history.

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Prepared by the **Slovenian Nuclear Safety Administration**

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URSJV/RP-085/2011

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Abbreviations

AB	Auxiliary Building
AC	Alternate Current
ADP	Administrative Procedure
AE	Accident Equipment
AEL	Additional Emergency Lighting System
AFW	Auxiliary Feedwater System
AFW TDP	AFW Turbine Driven Pump
ANS	American Nuclear Society
AOP	Abnormal Operating Procedure
ARP	Alarm Response Procedure
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
BTR	Boron Thermal Regeneration System
CAV	Cumulative Absolute Velocity
CB	Control Building
CCB	Component Cooling Building
CCW	Component Cooling Water System
CDF	Core Damage Frequency
CET	Core Exit Thermocouples
CI	Containment Spray System
CLA	Cask Loading Area
CPHRS	Civil Protection Headquarters of the Republic of Slovenia
CRDM	Control Rod Drive Mechanism
CSF	Critical Safety Functions
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
CVCS PDP	Positive Displacement Pump of the CVCS system
CY	Condensate System
DBF	Design Basis Flood
DC	Direct Current
DFC	Diagnostic Flowchart
DGB	Diesel Generator Building
DHR	Decay Heat Removal
DMSG	Decision Making Support Group
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EIP	Emergency Implementation Procedures
EMS	European Macroseismic Scale
ENSREG	European Nuclear Safety Regulator Group
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone
ERO	Emergency Response Organization
ESFAS	Engineered Safety Features Actuation System
ESD	Engineering Support Division
ESF	Engineered Safety Features

ESP	Engineering Services Procedures
ESW	Essential Service Water System
ETE	Evacuation Time Estimate
FHB	Fuel Handling Building
FP	Fire Protection System
FRG	Function Restoration Guideline
FRP	Fire Response Procedures
FTC	Fuel Transfer Canal
GPP	Gas Power Plant
HCLPF	High Confidence of Low Probability of Failure
HEPA	High Efficiency Particulate Air
HELB	High Energy Line Break
HPME	High Pressure Melt Ejection
HPP	Hydro Power Plant
HPSI	High Pressure Safety Injection
HRA	Human Reliability Assessment
HVAC	Heating, Ventilating and Air Conditioning
IA	Instrument Air
IAEA	International Atomic Energy Agency
IPE	Individual Plant Evaluation
IPEEE	Individual Plant Examination for External Events
LBLOCA	Large Break LOCA
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPSI	Low Pressure Safety Injection
m.a.A.s.l.	meters above Adriatic Sea level
MBLOCA	Medium Break LOCA
MCCI	Molten Core-Concrete Interaction
MCR	Main Control Room
MKSID	SNSA Communication System for Emergency Response (*Medresorni Komunikacijski Sistem med Izrednim Dogodkom*)
MOV	Motor Operated Valve
MW	Reactor Makeup Water System
NEK	The Krško NPP (*Nuklearna elektrarna Krško*)
NFPA	National Fire Protection Association
NNS	Non-Nuclear Safety
NPP	Nuclear Power Plant
NUREG	US Nuclear Regulatory Commission Regulation
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OLM	On-Line Maintenance
OSC	Operating Support Center
PARMS	Post-Accident Radiation Monitoring System
PASS	Post-Accident Sampling System
PCN	Process Computer Network
PET	Plant Evaluation Team
PGA	Peak Ground Acceleration
PIS	Process Information System
PMF	Probable Maximum Flood
PORV	Power Operated Relief Valve
PSA	Probabilistic Safety Assessment

PSHA	Probabilistic Seismic Hazard Analysis
PSRA	Probabilistic Seismic Response Analysis
PSR	Periodic Safety Review
PW	Pretreatment Water System
QA	Quality Assurance
RC	Release Categories
RCFC	Reactor Containment Fan Coolers
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RERP	Radiological Emergency Response Plan
RG	Regulatory Guide
RHR	Residual Heat Removal System
RMS	Radiation Monitoring System
RPV	Reactor Pressure Vessel
RTD	Resistance Temperature Detector
RTS	Reactor Trip System
TDS	Transformer Distribution Station
RWST	Refueling Water Storage Tank
SACRG	SAMG Control Room Guide
SAG	Severe Accident Guideline
SAME	Severe Accident Management Equipment
SAMG	Severe Accident Management Guidelines
SBLOCA	Small Break LOCA
SBO	Station Blackout
SCBA	Self-Contained Breathing Apparatus
SCG	Severe Challenge Guideline
SCST	Severe Challenge Status Tree
SFP	Spent Fuel Pool
SFPCCS	Spent Fuel Pool Cooling and Cleanup System
SG	Steam Generators
SI	Safety Injection
SLB	Steam Line Break
SNSA	Slovenian Nuclear Safety Administration
SOP	System Operation Procedure
SPSA	Seismic Probabilistic Safety Assessment
SSC	System, Structure and Component
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
TLD	Thermoluminescent Dosimeter
TSC	Technical Support Center
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
USAR	Updated Safety Analysis Report
US NRC	United States Nuclear Regulatory Commission
VCT	Volume Control Tank
WOG	Westinghouse Owners Group
WT	Water Treatment System

I Introduction

Slovenia, as the smallest nuclear country in the world, has only one nuclear power plant Krško with only one unit. It is a 2-loop Westinghouse designed nuclear power plant (NPP) with the net electrical output of up to 696 MWe. Its commercial operation started in 1983.

The Krško NPP is in the process of completion of its 1st Periodic Safety Review (PSR) action plan and at the beginning of the 2nd PSR. Likewise, the process of the plant design life time extension is on-going and it is expected to be concluded next year.

During almost 30 years of operation various safety reviews and improvements, upgrades and modernizations were performed. The most important examples from the past are plant modernization with power up-rate and steam generator replacement, Probabilistic Safety Analysis (PSA) related studies and upgrades (e.g. fire protection upgrade), adoption of Severe Accident Management Guidelines (SAMG), seismic reviews, analyses and upgrades (e.g. installation of the 3rd emergency diesel generator), wet reactor cavity, plant specific full scope simulator, etc.

After the Fukushima accident the operator of Krško NPP has performed its first and quick review trying to identify possible short-term improvements. In June 2011, based on the Krško NPP application, the Slovenian Nuclear Safety Administration (SNSA) licensed a series of minor modifications in the plant which add alternate possibilities for electrical power supply and cooling of reactor and spent fuel pool pool (SFP) in case of beyond design basis accidents (BDBA).

In response to the Fukushima accident, the SNSA issued a decision to the Krško NPP to perform a Special Safety Review. The programme of this review is completely in line with the ENSREG specifications for European Stress Tests.

The Krško NPP has fulfilled its commitment in time and sent the full scope Stress Test progress report to the SNSA by August 15. Likewise, the Final report was prepared and sent to the SNSA by the end of October 2011. The SNSA made a detailed review of the progress report and presented to the plant the findings and comments, which were taken into account in the Final report.

For the preparation of the stress test report the plant performed some additional analyses (e.g. evaluations of seismic and flooding margins, additional station blackout analyses to support the newest severe accident strategies, drain cycle of the 1E batteries, water heatup and evaporation rate in the spent fuel pool, evaluation of spent fuel pool criticality). These were all reviewed and approved by the technical support organizations with additional calculations (with different codes) done where appropriate. All these analyses and technical support organization reports were received by the SNSA.

Besides the stress test report, which covers the extreme natural conditions, the Krško NPP has also prepared an analysis regarding the impacts of aircraft crash on the plant. It shows that the plant is well designed and built, and with additional severe accident management equipment available onsite, well prepared even on such events. This report due to its sensitive nature is confidential and will remain as such.

In addition to obligate the plant to perform the stress tests, the SNSA also issued a decision requiring from the plant to reassess the Severe Accident Management strategy, existing design measures and procedures and implement necessary safety improvements for prevention of severe accidents and mitigation of its consequences. This evaluation shall be finished by January 2012. The action plan shall be reviewed and approved by the SNSA and completely implemented by the end of the year 2016.

Since the NPP Krško is the only NPP in Slovenia, it was decided that the contents of the Krško NPP's final report would be used as a basis of this national report and sent to the European Commission.

II Executive Summary

EARTHQUAKE

The Krško NPP is located in a seismically active region. At the time when the Krško NPP was designed and constructed the US NRC nuclear regulation and standards were used. Based on the Regulatory Guide (RG) 1.60 »Design Response Spectra for Seismic Design of Nuclear Power Plants, revision 1« the project acceleration of 0.3 g was used for Safe Shutdown Earthquake (SSE) and 0.15 g for Operating Basis Earthquake (OBE). The vertical component used is equal to the horizontal component in all frequency regions.

Regional geologic investigations for site selection began in the sixties. The location was later explored in detail with geomechanical, hydrogeological, geophysical and seismological investigations. These were performed in several stages. In the seventies the investigations included refractional measurements, soil survey, microseismical ground noise measurements, laboratory tests, gamma-gamma measurements, geoelectrical sounding of terrain, and density determination, all with the purpose to be used for geotechnical model of terrain evaluation and for the definition of the parameters of the earthquake effect. The Probabilistic Seismic Hazard Analysis (PSHA) made in 1994 increased Peak Ground Acceleration (PGA) to 0.42 g, while the 2004 PSHA study has further increased the seismic hazard to PGA of 0.56 g. The Seismic Probabilistic Safety Analyses (SPSA) finished in 1996 and 2004 were used to evaluate the plant's vulnerabilities to seismic events.

The NPP Krško Seismic Category I structures (e.g. containment vessel, shield building, interior concrete structures, control building, auxiliary building, intermediate building, essential service water intake and pump-house structure, diesel generator building and component cooling building) are dynamically analyzed for SSE (NRC RG 1.29) earthquake conditions using a modal analysis time history method. Safety Class 1, 2 and 3 systems are designated as Seismic Category I and a list of the related safety classification of equipment is provided. NPP Krško also complies with NRC Regulatory Guides, American Society of Mechanical Engineers (ASME), and the American Nuclear Society (ANS) codes in the area of piping, component and component support of safety related systems and components.

As part of the seismic PSA investigation, Individual Plant Examination for External Events (IPEEE) analysis for the seismic part was performed in the nineties (besides an Individual Plant Evaluation, IPE). That included a detailed walk-down of the plant to identify seismic vulnerabilities. The conclusion was that the plant had been well designed and constructed for a seismic event and no serious seismic issues were observed in containment. Also in the nineties a walk-down outside containment was performed, covering all components which were identified in the IPE as essential components for accident mitigation and safe shutdown of the plant. For all identified observations the Krško NPP performed appropriate corrective actions or design changes and resolved all deviations. In May and December 2003, a walk-down was conducted to assess new equipment added or replaced since 1996. In the 1995 SPSA, a fragility screening target of 2.0 g median capacity was set up to assure that any components screened out would have probability of a seismic induced failure at least two orders of magnitude less than the final predicted Core Damage Frequency (CDF). The new 2004 seismic hazard frequency has increased substantially. A new screening target has been set at 2.75 g median capacity with an associated High Confidence of Low Probability of Failure (HCLPF) value of about 1.0 g in order to assure the same probability of failure of screened out components relative to the expected final CDF.

The first periodic safety review represented a significant review process, where seismic issues were identified, evaluated, and new actions were set up for plant seismic improvements. One of the most important improvements will be the installation of a third seismically classified emergency diesel generator, which will be completed in 2012.

Seismic margins with weak points and cliff edge effects are evaluated first by means of identifying success paths from the safety analysis report and safety studies, then by mapping each critical safety function in every success path to the specific system, structure and component (SSC) with determining their seismic margins. The seismic margin for each critical safety function must be also determined. Critical functions for which the corresponding seismic margin was exceeded are assumed unavailable. In this manner all relevant cliff edges are identified. A success path becomes disabled when first of the required critical functions becomes unavailable. With increasing seismic severity, a number of success paths decreases. The point (seismic level) at which the last success path is disabled can be considered as the »seismic margin« for the whole plant.

Evaluation of seismic core damage margin, seismic margin for containment and spent fuel pool (SFP) integrity and cliff edge effects is presented. Seismic levels at which core damage would occur are considered to be at the PGA range of 0.8 g or higher. Seismic events at which early radioactivity releases to the environment would be likely to occur are considered to be of PGA significantly exceeding 1 g and late radioactivity releases in the range of 0.8 g to 0.9 g or higher. The SFP integrity would not be challenged for PGA's up to approximately 0.9 g. For earthquakes exceeding the PGA of 0.9 g, gross structural failures of SFP cannot be excluded and fuel uncovers are considered likely to occur.

Seismic events with PGA higher than 0.8 g were estimated to be very rare at the NEK site, with the return period is of the order of 50000 years or more.

Note that there is a difference between results from the progress and final report. These are due to corrections of the evaluations of seismic margins that were implemented on the basis of the technical support organization's recommendations.

FLOODING

The flooding section of the report systematically presents the description of the Krško NPP site that is located in an area prone to flooding, and the protection of the plant against external flooding. The Krško NPP is located in the Krško-Brežice Basin, on the left bank of the Sava river. The right bank of the Sava river above the Krško NPP and the left bank of the Sava river below the Krško NPP are extensive inundation areas that are flooded in events with high river flow.

The Krško NPP design basis flood is the 10000-year flood. The flooding protection was accomplished by the plant design and construction of the flood protection dikes along left banks of the Sava river and the Potočnica creek upstream and downstream of the plant. Plant building entrances and openings are constructed above the elevation of the 10000-year flood. The plant is protected also against the probable maximum flood with the appropriate design of the Sava river interface structures and with the flood protection dikes, provided that the greater quantities of water will flood the inundation on the right bank of the Sava river. Analysis of flash floods due to local heavy rainstorms is also described. The report presents in more detail the methodology, the input data and the results of analysis for the design basis flood. Considered is also the flood due to upstream dam's failure (seismic origin) and the effects of high wind on the raising of the water level. The report concludes that the plant design with additional flood protection dikes is adequate protection against the design basis flood.

The report provides an evaluation of external flooding margins at the plant. A range of flooding events, defined by the flooding flows and levels, is considered and for each range the success paths are defined as a minimum set of functions required to avoid the reactor core damage state and to preserve the containment integrity. The conclusions show that in case of an extreme flooding, much above the design flood or the probable maximum flood, the Krško NPP would be surrounded by water and thus would become an island. The flooding would not occur over the flood protection dikes along the Sava river, but from behind the NPP with the water coming from the flooding area on the left bank of the Sava river downstream of the NPP and the overflow of the Potočnica creek dikes.

The cliff edge effect is when a flood with a river flow 2.3 times larger than the design basis flood and 1.7 times larger than the existing probable maximum flood would flood the plant plain. Such a flood would have a return period of 1 million years. The challenges considered at probable maximum flood are loss of offsite power (due to overall conditions in the territory of Slovenia) or clogging of intake structures of the essential service water system (ESW). However, even in an extreme case with possible loss of emergency diesel generators, the Krško NPP provides strategies, personnel and equipment to be used with appropriate emergency operating procedures (EOP) and severe accident management guidelines (SAMG) that would prevent core damage and prevent or limit late releases.

The Krško NPP is in the process of upgrading its existing flood protection by raising the flood protection dikes upstream of the Krško NPP along the left bank of the Sava river and the Potočnica creek. After implementation of that modification the Krško NPP would not become isolated on an island even during the probable maximum flood.

The Krško NPP has also identified additional measures to increase robustness to external events and implemented them:

- Alternative means to provide suction to auxiliary feedwater system (AFW) pumps or to provide water to steam generators (SG) directly;
- Alternative means for power supply to chemical and volume control system (CVCS) positive displacement (PDP) charging pump in order to preserve reactor coolant system (RCS) inventory and integrity of reactor coolant pumps (RCP) seals in induced station blackout (SBO) or loss of essential service water system (ESW) / component cooling system (CCW) conditions;
- Alternative means for power supply to selected motor operated valves, as necessary for the implementation of alternative methods;
- Alternative means for providing water from the external sources to containment;
- Procedures for local operation of AFW turbine driven pump and for local depressurization by means of SG power operated relieve valves (PORV), both without need of DC or Instrument Power;
- Alternative means for makeup of SFP inventory.

LOSS OF ELECTRICAL POWER AND LOSS OF THE ULTIMATE HEAT SINK

The Krško NPP is connected to the 400 kV grid by one power line towards Maribor, and two towards Zagreb (Croatia). The switchyard 400 kV bus is also connected to the 110 kV system via 300 MVA transformers. For startup and emergency, the Krško NPP is also connected to the 110 kV power line Krško NPP – Gas power plant (GPP) Brestanica. Two unit transformers are connected between the generator load breaker and step-up transformers and provide normal onsite power supply for two Class 1E (MD1 and MD2) and two Non 1E (M1 and M2) 6.3 kV buses. If offsite power supply is lost, the two 6.3 kV emergency buses MD1 and MD2 are powered from their respective 3.5 MW emergency diesel generators. With the available fuel at the site, at least 7 days of emergency diesel generator operation is possible.

Each Class 1E train is provided with a complete 125 V DC system which supplies DC power to loads associated with the train. Each train’s system consists of a full capacity 125 V DC lead-acid 60 cell battery. The batteries are sized to supply DC loads for a minimum of four hours with a final discharge of 108 V (1.80 V per cell). The batteries have sufficient capacity per design to cope with a 4-hour station blackout (loss of all AC power), to provide safe shutdown of the unit. The capacity of each battery is 2080 Ah.

Among the components of the mobile equipment essential for managing severe accidents (Severe Accident Management Equipment, SAME) there are also 5 portable diesel generators. Establishing alternative power supply to the DC distribution panel and to the instrumentation distribution panels from portable diesel generators assures the long time availability of DC batteries and of 118 V AC instrumentation power supply (up to 72 hours with the fuel stored at the plant, or even longer if fuel would be supplied from offsite). For long-term operation, external support would be needed for diesel and gasoline supply to run the portable alternative equipment.

It was recognized that the core damage frequency for events initiated by loss of offsite power could be reduced by installation of the third emergency diesel generator. This modification is now in installation phase and is planned to be finished in 2012. The third emergency diesel generator will be located in a separate building with the third emergency bus which can be connected to either one of the existing emergency buses.

The essential service water system (ESW) provides cooling water to the component cooling system (CCW) and boron thermal regeneration system (BTR) to transfer the plant heat loads from these systems to the ultimate heat sink, the Sava river. The system also serves as a backup safety related source of water for feeding the steam generators through AFW. The ESW is classified as a Safety Class 3 and Seismic Category I system and is designed for operation with any water level varying from the original minimum river level, at an elevation of 147.85 m, to a maximum flood level at an elevation of 156.60 m. The temperature of the river water is considered to be a maximum of 26.7 °C and a minimum of 0.6 °C.

In a scenario with a loss of heat sink there is an assumption that the loss means the loss of connection between the pumps and the loads. All other systems operate normally and water is available from the Sava river. The envisaged alternate way of cooling the loads is by the existing fire protection system with demineralized water available onsite or from the Sava river. This alternative cooling has a limited cooling capacity, but would have enough capacity to allow operation of the centrifugal charging pump, high head safety injection pump and even the auxiliary feedwater pump or any other small heat load which would be necessary.

The report presents alternative possible cooling solutions. Alternative cooling could be established with the installation of T-connections on the existing ESW line to CCW heat exchangers to provide an alternative connection for fire protection pump with higher capacity. This could provide enough heat removal for one train of emergency core cooling system (ECCS) and also for removal of decay heat from the SFP. Installation of the connection on the non-safety related part of the piping and connection with portable fire protection pump could also provide alternative cooling for CCW heat exchanger. An alternative way of cooling the residual heat removal system (RHR) could be established with skid mounted pump and heat exchanger and connection points to the RHR system. A new water line from the Krško hydro power plant, which is planned to be installed in the near future, is also presented in the report. This will provide a passive way of cooling the RHR through the CCW heat exchangers using the gravity force to transport the cooling water.

SEVERE ACCIDENT MANAGEMENT

The severe accident management section of the report gives an exhaustive survey of measures taken in the Krško NPP in order to prevent the escalation of a reactor accident, mitigate the consequences of an accident involving severe damage of the nuclear fuel and to achieve a safe state of the reactor.

The decision-making process in case of an accident is well structured. The Krško NPP has in place a radiological emergency response plan (RERP), which is coordinated with RERP of local municipalities, region and with national RERP as well. Onsite emergency response organization is established and offsite support and assistance to the Krško NPP are provided by the local and other offsite support organizations.

Accident management measures for various stages of severe accidents are given in detail in plant's manuals and documents. The Krško NPP has in place upgraded emergency operating procedures (EOP) and severe accident management guidelines (SAMG), which provide adequate instructions for staff.

The plant is provided with equipment to manage an accident and emergency on site. The mobile equipment essential for managing a severe accident is stored on safe locations on site with respect to preventing their impairment in accident conditions. Fuel for at least 72 hours of operation of mobile equipment is also stored on safe location on site. All mechanical connections, tools, pumps required to implement severe accident management strategies are stored on site.

The Krško NPP has resources to manage the initial emergency response in case of a severe accident for the extended time up to 24 hours without any offsite support and up to one week with no needs for additional mobile equipment from offsite. The severe accident management equipment is regularly tested and maintained in accordance with plant procedures. The Krško NPP personnel is systematically trained in order to achieve good understanding of plant functioning and to be able to quickly respond to demanding tasks.

In the report the sensitivity analysis was performed to identify the risk, cliff edge effects and kinetics of severe accidents:

- The Krško NPP has a large dry containment. The hydrogen can be effectively handled using hydrogen control system and appropriate SAMG. No cliff edge effects were recognized concerning hydrogen induced threat to the containment fission product boundary.
- Potential failure of containment due to over pressurization can be significantly delayed by partial flooding of it. The failure of containment can be prevented (for 7 days) by spraying the containment using portable fire pump as an alternative.
- Damaged core has limited ability to return to critical conditions. However, injection of unborated water would lead to a controlled and stable reactor core state. Heat generated by critical core can be removed if sufficient water can be provided to refill the reactor vessel. No cliff edge effects related to core recriticality after core damage were recognized.
- No cliff edge effects were identified concerning the basement melt through due to wet cavity design and accident management measures preventing reactor vessel failure.
- The time margin to uncover the fuel assembly in SFP depends on the total heat power of the fuel elements inside the pool. If bounding case is considered, where reactor core has just been unloaded from the reactor it would take 3.2 days. More realistic but still conservative estimation is 11 days. In the SFP hydrogen production is not expected until the top of the fuel elements is uncovered, in bounding case this means 3.2 days.

Extensive destruction of the infrastructure, impairment of work performance due to high local dose rates and radioactive contamination are evaluated. Additionally, the unavailability of power supply, potential failure of instruments and potential effects from other neighboring installations at site are studied in detail.

The accident management measures for various stages of a severe accident are described for the nuclear power reactor and for SFP separately. The measures in case of loss of core cooling functions and installation design features for protecting containment integrity after occurrence of fuel damage are given. Possible actions for preventing fuel damage are studied and presented in the report.

Studies show that supply of the plant subsystems with electrical power is of utmost importance for nuclear safety. In accordance with this the Krško NPP is implementing additional safety upgrades. The measures include the installation of the third seismically classified emergency diesel generator (in progress) and acquiring another portable diesel generator (with four of them already on site). Also, flood protection will be improved (in progress) and new water pumping station will be acquired (in plans).

III Detailed Krško NPP report

1 General data about site/plant

1.1 Brief description of the site characteristic

The plant is located in the industrial zone on the northwestern brim of an alluvial valley surrounded by hills varying in relative elevation from 200 m to 700 m, east-southeast of the town Krško on the left bank of the Sava River in the Republic of Slovenia. The average altitude of surrounding area is about 154.5 meters above Adriatic Sea level (m.a.A.s.l.), while the plain with the nuclear power plant is located at the 155.20 m.a.A.s.l. The plant site represents an island slightly raised above the neighboring area.

The surrounding area of the site is sparsely populated. Except for a few small towns, Krško included, the area is mainly rural. In a 10 km circle, is located a population of 27700 inhabitants and in a circle of 25 km, are located 55000 inhabitants in Slovenia and 147700 inhabitants in Croatia.

Krško NPP owns the land within the site boundary in a rhomboid shape with sides approximately 400 m in length. Krško NPP has complete control of activities within a radius of 500 m from the plant centerline. The site boundary is marked by a fence.

The site has good transport connections: railway to the international railway line (1.08 km from the site) and road to the international road E94 (3 km from the site).

There is only one unit located on the site. The license holder for the nuclear power plant is Krško NPP - Nuklearna elektrarna Krško, d.o.o. (NEK, d.o.o.).

1.2 Main characteristics of the unit

The Nuclear Steam Supply System (NSSS) is a Westinghouse pressurized water reactor with two coolant loops. The NSSS and the design and fabrication of the initial cores with 16 × 16 fuel assemblies were supplied by Westinghouse Electric Corporation, while the replacement Steam Generators, type Siemens SG72W/D42 were provided in 1999 by the consortium Siemens-Framatome. The large dry containment of free volume of around 40000 m^3 consists of a cylindrical steel shell with a hemispherical dome and ellipsoidal bottom designed to accommodate normal operating loads, functional loads resulting from a loss-of-coolant accident, and the most severe loading predicted for seismic activity. A concrete shield building surrounds the steel shell to provide biological shielding for both normal and accident conditions and to provide collection and holdup for leakage from the containment vessel. Inside the containment structure, the reactor and other safety components are shielded with concrete. In addition to a containment spray system, a containment recirculation and cooling system is provided to remove post-accident heat. The reinforced concrete Reactor Building was designed by Gilbert Associates, Inc. The Krško plant is connected to the 400 kV national grid by three power lines, two of them toward Zagreb, and one toward Maribor. The switchyard 400 kV bus is also extended to transformer distribution station (TDS) Krško 400 kV bus and connected to the 110 kV system via 300 MVA transformer. For startup and emergency, the Krško Plant is also connected to the 110 kV grid by the power line Krško NPP - GPP Brestanica via TDS Krško.

The reactor is designed to operate at core power levels up to 1994 MWt, which corresponds to a net electrical output of approximately 683 MWe (considering the average weather conditions). First criticality was achieved in September 1981.

The fuel handling building is an integral part of the plant and is a reinforced concrete structure that utilizes shear walls and beam and slab floor systems. It is designed in accordance with the seismic and other criteria for safety structures. The spent fuel pool within the fuel handling building is lined with stainless steel to prevent leakage of water.

The plant is designed in accordance with the US NRC regulations and standards. All these standards and regulations are the basis for plant design and plant features as described in the Updated Safety Analysis Report (USAR). The plant is permanently upgraded and modernized in accordance with new industrial and regulatory requirements and standards. The Slovenian nuclear law, the »Ionising Radiation Protection and Nuclear Safety Act«, requires periodic safety reviews every ten years, which are the means for assessment of plant design, and condition of SSC, safety analyses, performance and feedback of experience, management, environment impact, licensing and regulatory requirements with latest standards, and good industry practices.

1.3 Significant safety differences between units

Not applicable as there is only one unit located at the site.

1.4 Scope and main results of Probabilistic Safety Assessments

The probabilistic safety assessment (PSA) analyses consist of two major parts: At-power PSA and Shutdown PSA. Different tools are used for power operation PSA, where more or less stable operation is expected in contrast to shutdown, where dynamic changes in plant status and configuration are conducted.

The at-power PSA consists of Level 1 – core damage frequency evaluations and Level 2 – release frequencies evaluation for large early release frequency (LERF) and other release categories.

Internal and external initiators are taken into risk account:

- Internal initiating events
- Internal fires
- Internal floods
- Seismic events
- High energy line brakes
- Other external events: high winds, aircraft accidents, external flooding, external fires and others.

The shutdown PSA focuses on plant states and configuration where risk is dominated by plant configuration status. On the following figures the living PSA results are presented for power operation as well as shut down conditions.

NPP Krško is constantly providing different activities which efforts increase safety and thus to reduce risk measured as core damage frequency and large early release frequency. Among those activities are plant modifications, procedure updates, following of current standards and methodologies and others. The most important contribution in increasing plant safety are plant modifications. The most important modifications implemented, that had large safety impact in reducing core damage frequency are presented in following description.

1.4.1 Plant modernization – Integrated Safety Assessment

A project »Integrated Safety Assessment of NPP Krško Modernization« was initiated to develop an updated PSA model and for use in the evaluation of risk.

Consistently with the concept described above the project tasks were divided in two stages:

Stage 1 addressed all plant modifications implemented since 1992. Within this stage, the modifications were evaluated with respect to their impact on the PSA model; the model was updated and quantified. Since then a regular update of the PSA master database is carried out to reflect the current plant operational experience. The Stage 1 results were compared with the integral NEK PSA Risk Spectrum model representing plant risk profile in 1992.

Stage 2 addressed another part of plant modifications like: changes that related to Steam generator replacement / power up-rate. These modifications were evaluated with respect to their impact on the PSA model.

In updating the PSA model a systematic review and screening process was followed to assure that the modifications which may influence the PSA are identified and addressed.

During the updating of the PSA, the PSA database was also updated incorporating the plant operating experience over the same period.

The Stage 1 revision to the NEK PSA involved substantial efforts which included modifications to many aspects of the PSA, as summarized below:

- Modification to the PSA model, mainly in system fault trees, to reflect the changes in the plant due to the modifications implemented,
- Modifications in the internal initiating event frequency reflecting the recent initiating event data,
- Modification to the PSA input data reflecting recent operating experience which incorporates impact of relevant modifications made,
- Incorporation of on-line maintenance (OLM) unavailability, based on data from approximately one year experience in such practice, into the PSA, and
- Modifications to the human error probability due to hardware and procedure modifications that affect human actions modeled in the PSA.

The total CDF obtained for the plant in the Stage 1 PSA update process is slightly lower than the previous estimate obtained in NEK previous model.

Significant changes in the plant risk profile are achieved through the modifications of Stage 2. The reduction is the combined effect of the many changes implemented as part of the Stage 1 and 2 modifications. Contribution of different initiator types to total CDF changed. Following the modifications, seismic events contribution is similar to internal events, and the internal fire became the lowest contributor to CDF. Positive changes in the Level 2 risk profile were also achieved (shift from large release toward smaller release categories).

Improvements are made in the plant through the modifications contributing to the reduction of plant risk.

Reduction in core damage frequency

The total NEK CDF is reduced to 1.28E-04 /yr from the previous estimate of 2.3E-04 /yr. This is a factor of 1.7 or 42% reduction.

The reduction in the total CDF is the combined effect of the many modifications implemented as part of the Stage 1 and 2 modifications. The main contributors to this reduction are SG replacement including associated changes, and implementation of the fire protection action plan; wire wrapping, installation of fire protected doors, additional sprinklers and other modifications.

Shift toward smaller releases

The plant internal event release categories (RC) frequencies contributing large releases have decreased while increasing RC frequencies with very small to small releases. This desirable shift is obtained largely due to SG replacement and »Wet Cavity« design. SG replacement significantly reduces RC frequencies (RC8A and RC8B) relating to bypass failure modes. The »Wet Cavity« design reduces RC frequencies (RC3B and RC5B) involving molten core-concrete attack while increasing those with no containment failure (RC2) and those involving no molten core-concrete attack (RC3A and RC5A).

Plant risk sensitivity is reduced or unchanged

Sensitivity analyses were carried out to assess if the plant risk was sensitive to any new aspects due to the modifications made. Sensitivity analyses did not identify any new components or aspects to which the plant is additionally sensitive following the modifications. Due to the reduction of plant vulnerabilities, as discussed above, many previously important component failure modes and operator actions were now eliminated.

No undue risk impact from plant operating practices

As part of the PSA modifications, the plant operating experience was used to update the PSA input parameters. Also, the practice of on-line maintenance introduced into the plant was included in the PSA model. The reduction in CDF and shift in release categories obtained include the impact of on-line maintenance. In other words, there is no undue impact of on-line maintenance or of other plant practices.

1.4.2 Fire protection action plan

As already mentioned in the chapter 1.4.1., the Fire protection action plan prioritized proposed fire protection modifications contained in Krško NPP Fire Hazards Analysis - Safe Shutdown Separation Analysis, the International Commission for an Independent Safety Assessment Analysis of Core Damage Frequency Due to Fire at the Krško Nuclear Power Plant, and the Operational Safety Review Team reports that used a risk-based approach which will provide reduction of the probabilistically significant contributors to fire-induced core damage frequency.

The most important modifications to the plant that reduced overall fire-induced risk were:

1. Circuit isolation of vital equipment located on the evacuation panels and motor control centers and the addition of reactor coolant system wide-range temperature and power-operated relief valve control at the evacuation panels;
2. Rerouting train B cables outside of Fire Area CB-3A;
3. Installation of a sprinkler system above train B cables and fire wrapping of some train B cables in the auxiliary building basement; and
4. Installation of a sprinkler system above and heat shield between the essential service water system pumps, and fire wrapping of an essential service water system pump power cable.

Implementation of these modifications alone reduced the fire-induced core damage frequency from 9.71E-5 /ry to 1.24E-5 /ry.

1.4.3 Impact of instrument air (IA) modification on safety

NEK Modification 390-IA-L introduced replacement of two IA Compressors: IA904CPR-001 and IA904CPR-002. IA Compressors replacement impacted Instrument Air system (and partially Component Cooling system) in the following points:

- Compressors 1 and 2 are now 100% not 50% as were before (each compressor capable of supplying both trains; before, each compressor supplied one train);
- Some IA piping was changed, where compressors are connected to the IA trains;
- Component cooling piping was changed, because old compressors needed controlled cooling pressure, but the new compressors can be connected directly to the CC pressure.

From PSA point of view, this modification had positive safety impact. The internal initiating events CDF decreased by 5.4% which in terms of total CDF comes to 1.5% of the total CDF value.

1.4.4 Updated human reliability model

The most important changes that were introduced in the PSA model as the result of human reliability assessment (HRA) update were new human error probabilities of human failure events. These human error probabilities included consideration of dependencies between human failure events.

On-site full scope control room simulator was used in the evaluation and was accounted as a tool for operator training.

Identification of the most important human failure events from the lists of risk factors such as risk increase factor and risk decrease factor was performed.

The results of evaluation of the current Krško NPP PSA model considering human error probabilities for human failure events showed that contribution of human operators to the plant risk is an important contribution, which has to be appropriately addressed at least as much as it has been considered up to now.

CDF for internal initiating events was calculated using computer code Risk Spectrum. HRA update results in reduction of CDF for internal initiating events for approximately 2%.

1.4.5 Seismic probabilistic safety assessment

A seismic probabilistic safety assessment (SPSA) was conducted for the Krško NPP in 1996. SPSA involved the integration of three separate engineering disciplines: (1) the development of the seismic hazard, (2) the development of event tree/fault tree risk models of the plant response to earthquake induced transients and failures, and (3) development of seismic fragilities of structures and components. The seismic hazard was defined as the frequency of occurrence of peak ground acceleration (PGA) and the fragilities were defined as conditional probability of failure versus PGA. The development of fragilities also takes into account the shape of the ground motion spectrum that is predicted in the seismic hazard studies. The integration of the earthquake PGA frequency and the structural and component fragilities resulted in unconditional seismic induced failure rates of the structures and components. These failure rates were propagated through the event tree/fault tree risk model and combined with random failures and human error to quantify the core damage frequency (CDF) and large early release frequency (LERF).

The frequency of the PGA and the ground motion response spectrum shape were defined in a probabilistic seismic hazard analysis (PSHA) conducted by teams of earth scientists, seismologists and engineers and were used in the SPSA. As part of the first periodic safety review, a new PSHA was conducted in 2002 and revised in 2004 taking into account more recent geologic, seismologic, geophysical and geodetic investigations. The new seismic hazard was more severe than the hazard used in the original SPSA. The frequency of occurrence of the PGA has increased by about a factor of two, but is offset to a small extent by a lower amplification in the ground motion spectrum. In further support of the periodic safety review, a study was conducted to compare the original seismic design of Krško NPP to current standards. These comparisons were reported and recommendations for additional evaluations were made. It was suggested that the most feasible approach was to update the SPSA to include the effects of the new seismic hazard and to include additional evaluations of SSCs.

A new ANS standard has been developed for external event Seismic Probabilistic Risk Assessments (SPRA), (ANS, 2003), that provided requirements for three capability categories or levels of detail for SPRA.

In conducting the update of the Krško SPSA, it was desired to meet the requirements of Capability Category II of the ANS Standard. Capability Category II requires that the performance of relays be explicitly modeled and that a full uncertainty analysis be conducted.

Several preliminary studies were conducted that lead up to the full update of the Krško SPSA. A feasibility study was conducted to determine the effect of the new seismic hazard using the existing SPSA model. As expected, the predicted CDF increased by a factor of 2. A number of parametric studies were then conducted to determine the effects of increased capacity of selected SSCs such as loss of offsite power, SBLOCA, inclusion of a small 0.6 kVA portable 400 V diesel generator, increasing the seismic fragility screening level, upgrading selected components, etc. It appeared feasible to decrease the calculated CDF by more than a factor of 2, thus returning the calculated CDF to a value near that of the original SPSA.

The scope of update was to document the update of existing seismic fragilities, the addition of new fragilities and to specify changes to be made in the SPSA model. The steps involved in this activity were:

- Perform a walk down to assess the increased capacity of components that were upgraded as a result of the 1996 SPSA.
- Scale existing probabilistic spectra to reflect the change in the shape of the ground motion spectral shape.
- Revise existing fragilities to account for the scaled probabilistic spectra and results of the walk down.
- Develop new fragilities for components replaced or added since the 1996 SPSA.
- Develop new fragilities for selected subsystems where generic capacities were previously used (SBLOCA, MBLOCA, LBLOCA).
- Develop revised fragilities for components and commodities previously screened out (electrical raceways, buried piping, etc.)
- Perform a comprehensive relay evaluation and determine the consequences of seismic induced relay malfunction.
- Specify changes to be made to the existing SPSA model to incorporate relay performance, additional components and additional human actions.

Major modifications that were performed to improve seismic safety were installation of cable supports in control room ceiling, additional supports to water tanks, additional supports for some installations and air ducts, welding and anchoring of electrical control racks, additional penetration sealing and others.

While the increase in seismic hazard was expected to increase the calculated CDF, changes in the SPSA model accounting for new equipment added to enhance safety, taking into account the seismic capacities of new equipment that replaced existing equipment, taking credit for some systems previously assumed to be unavailable after a seismic event and removal of conservatisms in the plant model, had a reducing effect on the calculated CDF. Around 18% reduction of total CDF was accomplished.

1.4.6 Modification in progress – third diesel generator

Seismic PSA model was used to identify potential modifications to systems and to procedures that could result in cost-effective risk reduction. Results from the NEK PSA Model were needed in order to identify the dominant failure contributors, and then to identify potential modifications. However, a few potential modifications were initially identified for evaluation:

- Small 0.6 kVA portable 400 V diesel generator (risk reduction depends heavily on time);
- Condensate storage tank (CST) alternatives: build basin around CST, water tank truck;
- Enable feed and bleed (N2 tanks for valves);

- ESW alternative;
- Addition of a third main 6.3 kV diesel generator;
- DC powered letdown and reactor coolant pump (RCP) seal return isolation valves;

At the end addition of a third main 6.3 kV diesel generator was chosen for implementation. Although it is a major modification, the addition of a third large 6.3 kV diesel generator will have significant impact on plant seismic risk. It would also impact internal initiating event CDF. Around 35% reduction of total CDF is expected.

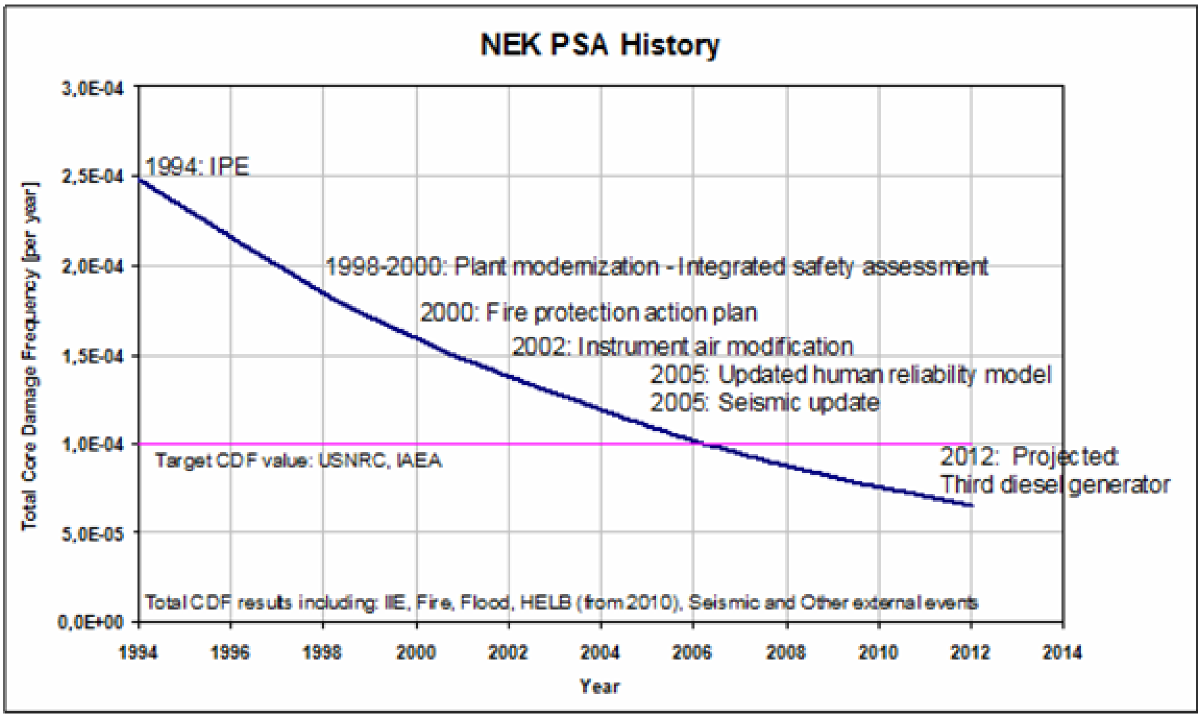


Figure 1: At-power total core damage frequency history [CDF /yr]

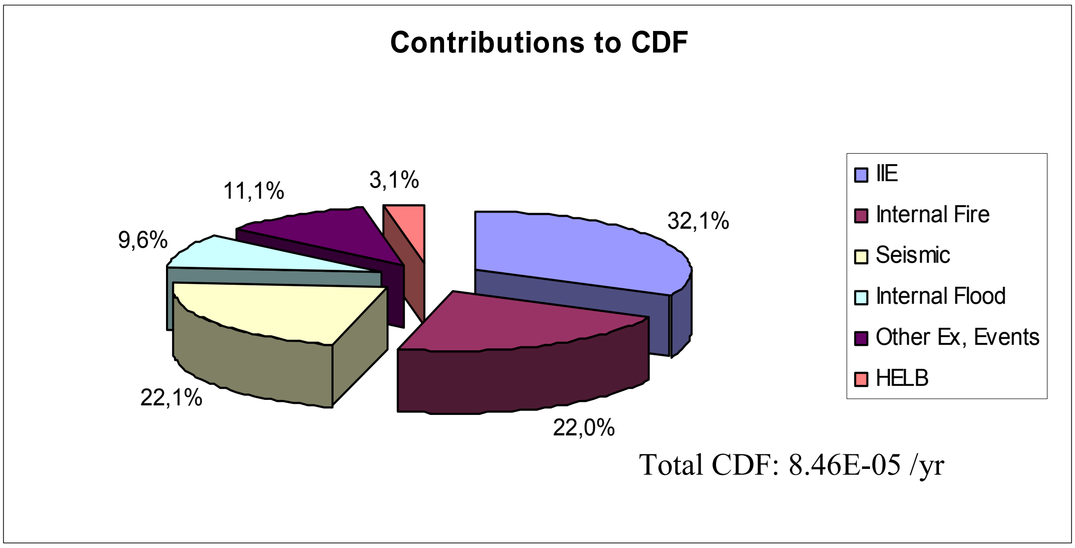


Figure 2: At-power contributions to total core damage frequency

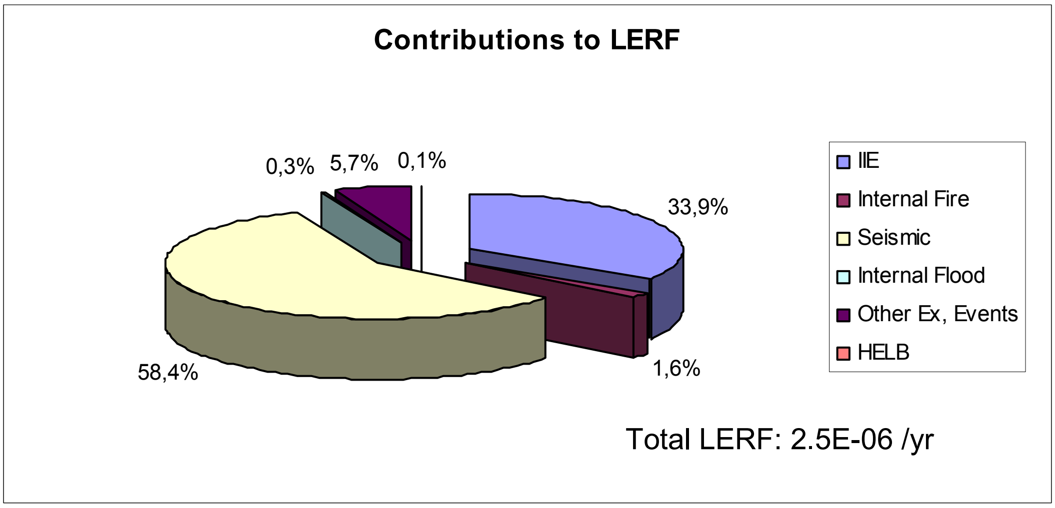


Figure 3: At-power contributions to large early release frequency

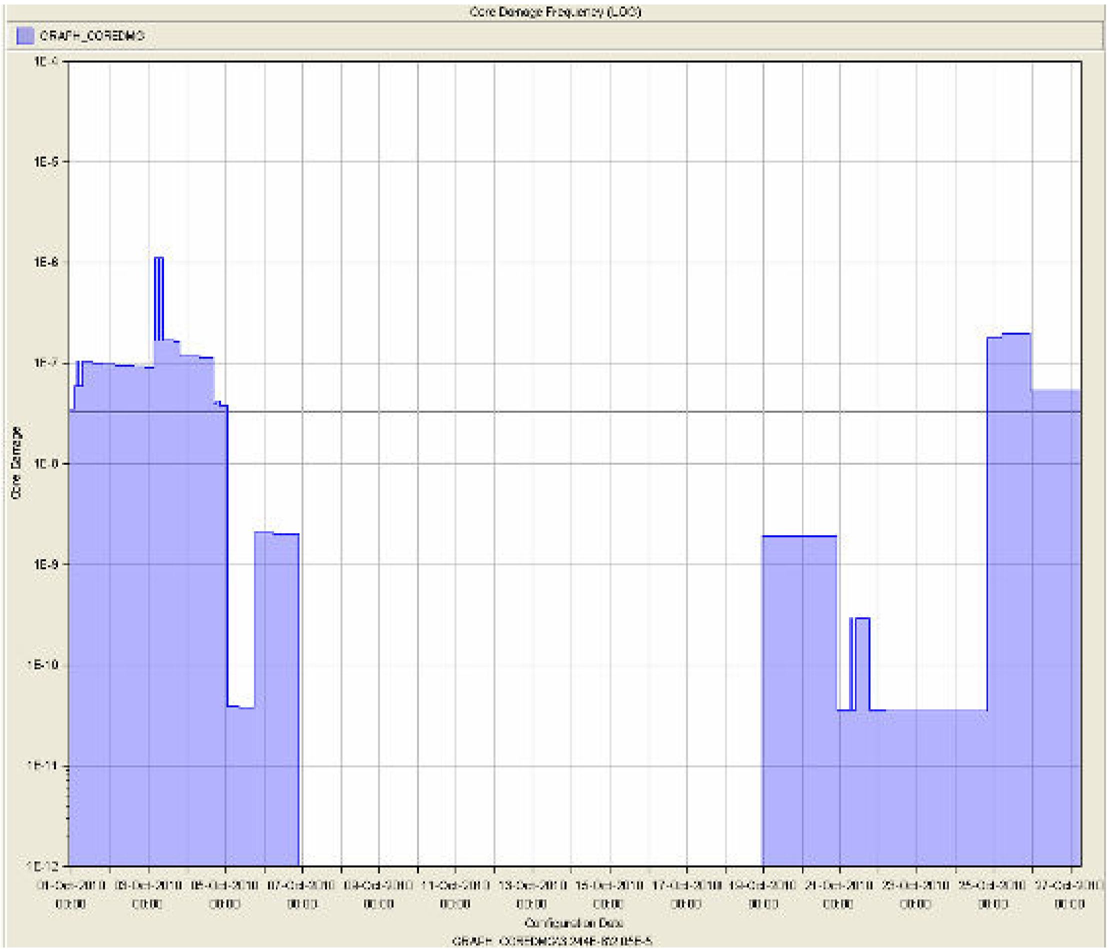


Figure 4: Shutdown core damage frequency profile
(Average CDF = 3.24E-08 /hr and core damage probability for the time of shutdown CDP = 2.05E-05)

2 Earthquake

2.1 Design basis

2.1.1 Earthquake against which the plant is designed

2.1.1.1 Characteristics of the design basis earthquake

Krško NPP is located in a region with moderate seismic activity. The main seismic-tectonic studies were performed before the plant construction from 1964 to 1968 and intensive studies continued in early seventies (1971 to 1975). The US NRC nuclear legislation was used at the time of Krško NPP design and construction, since the official domestic legislation (ex Yugoslavian technical standards) was not developed for nuclear installations. Based on Regulatory Guide 1.60 the project acceleration 0.3 g was used for Safe Shutdown Earthquake (SSE) and 0.15 g for Operating Basis Earthquake (OBE). For the operational basis earthquake loading condition, the plant is designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures, systems and components (SSC) are required to operate within design limits. The seismic design for the SSE is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that required critical structures and components do not lose their capability to perform their safety function. Not all critical components have the same functional safety requirements. For example, the reactor containment must retain capability to restrict leakage to an acceptable level. Therefore, based on present practice, general elastic behavior of this structure under the SSE loading condition must be ensured. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain their contents and allow fluid flow.

Design ground response spectra for the horizontal component SSE comply with NRC Regulatory Guide 1.60, »Design response spectra for seismic design of nuclear power plants, revision 1«. The vertical component used is equal to the horizontal component in all frequency regions.

Seismic design is derived from seismic input as design response spectra used for SSE and OBE peak ground accelerations (PGAs). The ground acceleration response spectra derived from synthetic earthquake time motion records was compatible with NRC Regulatory Guide 1.60, with a peak horizontal ground acceleration normalized with 0.3 g and vertical component used equal to the horizontal component in all frequency regions. The seismic system analysis of Seismic Structures Category I is performed with dynamic analysis using time history method on a modal model. The model consisted of mass points and stiffness elements composed in seismic models. The soil structure interaction is considered in this model since the PGA is valid for free field only and not for the foundation level. The resulted time history analyses are used to develop response spectra with different dumping values.

2.1.1.2 Methodology used to evaluate the design basis earthquake

The Krško NPP is located along the Sava river in the Krško basin in south-east Slovenia about 2 km from the town of Krško and about 40 km from city of Zagreb, Croatia. Regional geologic investigations for site selection began in 1964; the first detailed investigations of the Krško NPP site were conducted primarily between 1971 and 1975. Construction began in 1975; testing started in 1981; and the Krško NPP has been in commercial operation since 1983.

Detailed research work for geomechanical, hydro-geological, geophysical and engineering seismological investigations was carried out on the site itself. It was performed in several stages.

The first stage of the investigation covered the period 1971, 1972 and 1973 and included boring of the site up to 12-13 m in depth, refractional measurements of P and S wave velocities, geoelectrical trial boring of the terrain, gravimetric soil survey, and micro-seismic ground noise measurement. These investigations were carried out in the wider surroundings of the NPP and used for evaluation of its suitability and in the selection of the final location.

The second stage of the investigation was carried out in the second half of 1973 according to the IEEEs program. It covered seismic refraction measurements of P and S wave velocities and microseismic ground noise measurements at the NEK site. These investigations were used for the geotechnical model of terrain evaluation as well as for the definition of the parameters of the earthquake effect.

The third stage was carried out in the middle of 1974. This stage included 30 geomechanical borings of 30 – 90 m in depth, laboratory material tests, and distribution of seismic P and S wave velocities according to cross-hole methodology up to 45 m in depth.

The fourth stage was carried out at the end of 1974 as a supplementary investigation. It covered 24 new geomechanical borings with additional laboratory tests, measurements of seismic P and S wave velocities according to cross-hole methodology up to 100 m in depth, refractional seismic measurements of P and S wave velocities, and gamma-gamma measurements of material density and geoelectrical sounding of the terrain. In addition to the above mentioned investigation, six trial pits to 4 m in depth were excavated for relative density determination.

Investigations covering the second, third and the fourth stage were carried out at the NPP site and refer in general to structures of Category I.

The most recent stage of investigations began in 1991 when the question of seismic hazard at the Krško NPP site was posed in the Slovenian parliament. To answer this question, an »ad hoc« commission was formed in 1992, and its major findings were presented in a report that was partly published by Lukacs et al. The commission's conclusion was that additional investigations in the vicinity of the Krško NPP were needed.

In accordance with IPE (GL 88-20) and a licensing amendment imposed by the Slovenian Nuclear Safety Administration (SNSA), NPP decided to complete a Seismic Probabilistic Safety Analysis (SPSA) to evaluate the plant's vulnerabilities to seismic events. The first step in the SPSA, which began in 1991, was to develop seismic source models for the Krško region in order to develop probabilistic estimates of the free-field ground motions and uniform hazard response spectra. This work, which was done under the direction of the Faculty for Civil Engineering and Geodesy in Ljubljana, was published in 1993 and 1994. This study included both regional (150 km radius) and near-regional (25 km radius) seismic source models developed by three independent Earth Science teams. The results of this analysis indicate that the cumulative contribution of the regional sources constitutes less than 2% of the seismic hazard at Krško NPP. Based on statements by the seismic hazard experts and on further recommendations by the International Atomic Energy Agency (IAEA), it was concluded that there was insufficient information regarding local faulting. Therefore, SNSA and NPP decided to implement a phased program of geologic, seismologic, geophysical, and geodetic investigations.

Geologic, seismologic, and geophysical investigations were partly completed by local experts during 1994 to 1996. An enhanced seismic network began operation in 2002. Based on the preliminary results and on recommendations by the IAEA, the »Program for additional site investigations« was revised, and additional geologic, seismologic and geophysical investigations were performed. These studies, which focused on the site and near region (25 km radius), included the following:

- Update of the seismicity database.
- Detailed geologic mapping at a scale of 1:5000 of the Krško basin and adjacent regions in the vicinity of the site.
- Geophysical investigations, including the acquisition of: (1) four near-regional reflection lines totaling 55 km in length recorded by using an explosive source in 6 to 11 m-deep boreholes; (2) three near-regional lines totaling 9.5 km in length recorded using a

Hydrapulse source; and (3) six high-resolution profiles totaling 4.6 km in length across selected features.

- Detailed investigations of the Quaternary deposits, soils, and geomorphic surfaces that could be used to evaluate neotectonic deformation.
- Acquisition and analysis of geodesic leveling and GPS survey data.

The results of the geologic, seismologic, and geophysical investigations are presented in technical reports develop as a part of the first periodic safety review (PSR), which presents an updated seismotectonic model of the Krško basin. This report provided the basis for reevaluating and revising the Probabilistic Seismic Hazard Analysis (PSHA, 2002-2004) to provide the ground motion inputs for the SPSA.

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

As mentioned, the original design PGA of the SSE ground motion amounted to 0.3 g and it was applied without any reduction at the level of foundation. The soil-structure interaction in the original structural model was modeled by using spring elements representing soil.

The NPP developed a PSHA analyses in 1994. The resulting hazard determined within this study was used as an input for SPSA calculations. The results were valid for free field and direct comparison with design spectra were recalculated due to the known effect of soil-structure interaction. In a probabilistic response analysis, the characteristics of the free-field ground motion are defined by the shape of the median uniform hazard spectrum corresponding to a return period of interest. For the NPP, the uniform hazard spectrum shape corresponding to a 10000 year return period was used. This uniform hazard spectrum was developed from a hazard analysis of the site. The seismic level of the analyses was twice the SSE level, since the SSE level for NPP is 0.3 g and median uniform hazard spectrum for the probabilistic analyses were anchored to PGA of 0.6 g. The probabilistic analysis was carried out at an earthquake level equal to twice the design earthquake level. The response quantities of interest recovered from the multiple earthquake simulations included peak accelerations, maximum member forces, and floor acceleration time histories. These quantities were needed for fragility development. For the three structures analyzed, the median centered spectral peaks occur at lower frequencies when compared to the USAR results. The frequency shift can be easily explained as follows. Since the probabilistic analysis was performed at a higher earthquake level than the USAR analysis, the median soil stiffness properties for the probabilistic case are lower than those for the USAR case. In average, the ratio between the soil shear wave velocities is of the order of 0.6. Since the responses of these structures are mainly controlled by the soil, a shift on the dominant frequency of the responses of about 0.6 is also expected.

According to the PSHA study, completed in 1994, a larger PGA of 0.42 g can be expected at the surface. In the years 2002 to 2004 a revised PSHA was performed for Krško NPP site. The resulting frequency spectra of accelerations were calculated from the history of earthquakes in the region and from the activity of faults near the location of NPP. Different data was evaluated by independent groups of experts to gain a common probability and intensity of probable maximum acceleration at the location of the NPP. The aim of additional studies was to evaluate the vicinity of the NPP with total of 45 km seismic profiles.

This PSHA has further increased the seismic hazard to a PGA of 0.56 g. Based on additional analyses using new seismic hazard data and a more advanced realistic model for soil structure interaction, it was concluded that the peaks in the floor response spectra corresponding to PGA of 0.6 g, i.e. twice original design value, are similar to those obtained in the original design. This finding suggested that the Krško NPP can accommodate a ground motion of much higher intensity than it was designed for.

2.1.2 Provisions to protect the plant against the design basis earthquake

2.1.2.1 Key structures, systems and components required for achieving safe shutdown state and supposed to remain available after the earthquake

The design criteria used during design of Krško NPP and adopted for SSCs depend on the magnitude and the probability of occurrence of natural phenomena at the site. The designs are based on the most severe of the natural phenomena reported for the site with an appropriate margin to account for uncertainties in the historical data. Detailed discussion of the earthquake phenomena considered and the design criteria developed are presented in the Sections listed below. The design criteria developed meet the requirements of Criterion 2 of 10 CFR 50; Appendix A (Criterion 2 - Design bases for protection against natural phenomena).

2.1.2.2 Non-Nuclear systems and components

Based on the original Design Specifications the equipment which is not safety related but could impact the operability of the safety related equipment is designed and erected as Seismic Category I components. Such design ensures the structural integrity of those components during seismic events, no collapse, derail or drop its load as a result of the SSE. In accordance with plant procedures the same principle is used during execution of design modifications being constitutional part of continuous plant enhancement process.

SEISMIC CLASSIFICATION

The plant structures, the Engineered Safety Features (ESF) and other safety related systems and components, are identified and classified in accordance with the requirements of General Design Criterion 2 of Appendix A to Title 10 CFR Part 50, General design criteria for nuclear power plants, and Appendix A to Title 10 CFR Part 100, Seismic and geologic siting criteria for nuclear power plants. NRC Regulatory Guide 1.29 designates those SSCs which must be designed to remain functional during the SSE as Seismic Category I. Specifically, if a SSE occurs, all Seismic Category I SSCs must withstand the effects of the SSE and assure:

- The integrity of the reactor coolant pressure boundary;
- The capability to shut down the reactor and maintain it in a safe shutdown condition; and
- The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of Title 10 CFR Part 100.

Seismic Category I structures

Seismic Category I structures typically include those classified by the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) as Safety Classes 1, 2, and 3 (i.e., safety related).

NPP Seismic Category I Structures are listed below:

- Containment vessel
- Shield building
- Interior concrete structures
- Control building
- Auxiliary building
- Fuel handling building
- Intermediate building

- Essential service water intake and pump-house structure
- Diesel generator building
- Component cooling building

Seismic Category I mechanical components and systems

Mechanical components and systems are classified according to ANSI N18.2-1973 Safety Classes, and are listed below:

REACTOR COOLANT SYSTEM

- Reactor Vessel
- Full Length CRDM Housing
- Part Length CRDM Housing
- Steam Generator
- Pressurizer
- Reactor Coolant Hot and Cold Leg
- Piping, Fittings & Fabrication for Safety Class for other piping and associated valves in the Reactor Coolant System and other Auxiliary Systems,
- Surge Pipe, Fittings & Fabrication
- Loop Bypass Line
- RTD Bypass Manifold
- Safety Valves
- Power Operated Relief Valves
- Reactor Coolant Pump

CHEMICAL & VOLUME CONTROL SYSTEM (partially)

- Regenerative Heat Exchanger
- Letdown Heat Exchanger
- Mixed Bed Demineralizer
- Cation Bed Demineralizer
- Reactor Coolant Filter
- Volume Control Tank
- Centrifugal Charging Pump
- Positive Displacement Pump
- Seal Water Injection Filter
- Letdown Orifices
- Excess Letdown Heat Exchanger
- Seal Water Return Filter
- Seal Water Heat Exchanger
- Boric Acid Tanks
- Boric Acid Transfer Pump
- Boron Thermal Regeneration Subsystem
- Moderating Heat Exchanger
- Letdown Chiller Heat Exchanger
- Letdown Reheat Heat Exchanger
- Thermal Regeneration Demineralizer

EMERGENCY CORE COOLING SYSTEM

- Refueling Water Storage Tank
- Accumulators
- High Head Safety Injection Pumps
- Boron Injection Tank (discharge)
- Boron Injection Tank Recirculation Pump
- Boron Injection Surge Tank
- Boron Injection Flush Orifice

RESIDUAL HEAT REMOVAL SYSTEM

- Residual Heat Removal Pump
- Residual Heat Exchanger

BORON RECYCLE SYSTEM (partially)

- Recycle Holdup Tank
- Recycle Evap. Feed Pump
- Recycle Evap. Feed Demineralizer
- Recycle Evap. Feed Filter
- Recycle Evap. Condensate Demineralizer
- Recycle Holdup Tank Vent Ejector
- Recycle Evaporator

REACTOR MAKEUP WATER SYSTEM

- Reactor Makeup Water Pipe
- Reactor Makeup Water Pump
- Reactor Makeup Water Storage Tank

WASTE PROCESSING SYSTEM (partially)

<u>(LIQUID SUBSYSTEM)</u>

- Reactor Coolant Drain Tank Heat Exchanger
- Waste Holdup Tank
- Waste Evaporator Feed Pump
- Waste Evaporator Feed Filter
- Waste Evaporator

<u>(GAS SUBSYSTEM)</u>

- Gas Decay Tanks
- Hydrogen Recombiner - Catalytic
- Gas Compressor

FUEL HANDLING SYSTEM

- Reactor Vessel Head Lifting Device & Portions that furnish support to Control Rod Drive Mechanisms
- Spent Fuel Pool Bridge & Hoist
- Rod Cluster Control Changing Fixture
- Fuel Transfer System
- Fuel Transfer Tube & Flange

– Remainder of System

OTHER AUXILIARY SYSTEMS

COMPONENT COOLING SYSTEM

– Component Cooling Heat Exchanger
– Component Cooling Pump
– Component Cooling Surge Tank
– Piping
– Valves

ESSENTIAL SERVICE WATER SYSTEM

– Essential Service Water Pumps
– Essential Service Water Piping
– Essential Service Water Screens
– Essential Service Water Valves
– Essential Service Water Strainers
– Essential Service Water Gates
– Essential Service Water Trash Rakes
– Essential Service Water Screen Wash Pumps

REFUELING WATER STORAGE SYSTEM

– Refueling Water Storage Tank
– Piping
– Valves

CONTAINMENT SPRAY SYSTEM

– Pumps
– Piping
– Valves
– Spray Nozzles

REACTOR VESSEL OR CORE RELATED

– Reactor Vessel Support Shoes and Shims
– Reactor Vessel Head and Shell
– Insulation
– Reactor Vessel Internals
– Control Rod Guide Tubes
– Control Rod Drive Mechanism Assemblies

INCORE INSTRUMENTATION (MECHANICAL)

– Thimble Guide Tubing
– Thimble Seal Table and Parts
– Thimble Guide Couplings
– Flux Thimble Assembly

SPENT FUEL POOL COOLING SYSTEM

– Heat Exchanger

- Piping
- Pumps

NUCLEAR SAMPLING SYSTEM

- Piping and Valves
- Sample Heat Exchanger

AUXILIARY FEEDWATER SYSTEM

- Auxiliary Feedwater Pumps
    i. Electric Motor Driven
    ii. Steam Turbine Driven
- Piping and Valves
- Condensate Storage Tank

DIESEL GENERATOR

- Diesel Oil Tank
- Diesel Oil Pump
- Piping from D.F.O. Storage Tank to the Diesel Generator
- Valves

INSTRUMENT AND STATION AIR SYSTEM

- Piping and Valves

MAIN STEAM SYSTEM

- Main Steam Piping
- Steam Generator to Isolation Valves
- Main Steam Isolation Valves
- Steam Generator Safety Valves
- Steam Generator Relief Valves

STEAM GENERATOR BLOWDOWN SAMPLING SYSTEM

- Steam Generator Blowdown Sample Cooler
- Steam Generator Blowdown Sample Pipe

STEAM GENERATOR BLOWDOWN SYSTEM

- Steam Generator Blowdown Piping

FEEDWATER ISOLATION VALVES

- Automatic Check Valves and
- Air Operated Gate Valves
- Piping from Isolation Valves to the Steam Generators

HEATING, VENTILATING AND AIR CONDITIONING SYSTEMS (HVAC)

- Containment Recirculation Fans &Hydrogen Control
- Hydrogen Recombiner - Electric 2
- Hydrogen Purge Fans
- Annulus Ventilation Fans
- Annulus Exhaust Filter Plenum

- Main Control Room Air Handling
- Main Control Room HVAC Plenums
- Computer, CRDM Control and
- Switchgear Room HVAC AHU
- Main Control Room Emergency Recirculation Charcoal System
- Main Control Room Emergency Recirculation Charcoal System Filter Plenums
- Battery Room Ventilation AHU
- Safety Injection, Residual Heat
- Removal and Spray Pump Rooms Air Handling Units
- Heat Exchanger and Pump Rooms, and Spent Fuel Pool Area Charcoal Exhaust Fans
- Spent Fuel Pool Area Charcoal Exhaust Plenums
- Spent Fuel Pool Emergency Supply Air System Fans
- Chilled Water Systems
- Essential Services Pump House Fans
- Diesel Generator Room Ventilation Fans
- Component Cooling Building Ventilation Fans

ELECTRICAL EQUIPMENT

- Motors 1E
- Control Panels (for Main Equip. Control Cubicles)
- Switchgear Including Metal-Enclosed Bus
- Protectors
- Motor Control Centers
- Storage Batteries and Racks
- Battery Chargers
- Inverters
- Diesel Generators
- Electrical Penetrations
- AC and DC Instrument and Power
- Lighting Panels and Transformers
- Breakers and Circuit
- Heat Tracing Cable

I&C SYSTEMS

- NIS
- ICCMS
- Reactor Protection System
- Control room cabinets and panels

Summary on Category I SSCs:

The containment vessel is a Seismic Category I structure with an internal free air volume of 40000 m^3. It is designed for a maximum internal pressure of 0.31 MPa rel. with a coincident temperature of 128 °C under accident conditions and a maximum external pressure differential of approximately 0.01 MPa due to accidental operation of the Containment spray system (CI). Design of the containment vessel considers dead load, live load, construction loads, temperature gradients and the effects of penetrations for accident conditions (including seismic considerations) as well as normal operating conditions.

All Class 1 mechanical components and supports are designed and analyzed for the Design, Normal, Upset, and Emergency Conditions to the rules and requirements of the ASME Code Section III. The design analysis or test methods and associated stress or load allowable limits that have been used in evaluation of Faulted Conditions are those that are defined in ASME Code with supplementary methods outlined below:

1. Elastic system analysis and component inelastic analysis
2. Elastic / inelastic system analysis and component / Test load method
3. Component support buckling allowable load

Loading combinations and allowable stresses for ASME III Class 1 components and supports are provided for Faulted condition, the effects of the SSE and postulated piping ruptures are combined using the square root of the sum of the squares method.

Using the principle of seismic design Class I SSCs it is assured that the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition; and the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of Title 10 CFR Part 100, are preserved.

2.1.2.3 Main associated design/construction provisions

The control point for defining the seismic ground motion is specified to be at the ground surface. Damping is stated to be either, RG 1.61 values or values contained in ASME B&PVC Appendix N, which would allow 5% damping for piping systems, rather than the lower RG 1.61 values used in the NPP design.

The control point for the NPP SSE was assumed to be at the basement of the structures, thus for embedded structures, there is considerable conservatism in the input motion. This conservatism, coupled with the use of soil spring modeling and time dependent analysis, results in very conservative response spectra that define the input to equipment and subsystems mounted in the structures.

The NPP seismic design criteria specified that the peak value of the response spectrum could be used for single degree of freedom systems. For systems that could respond in multiple modes of vibration, 1.5 times the peak of the response spectrum was used.

Simplified piping analysis was conducted by assuming that single pipe spans are simply supported. An equivalent static load is calculated for piping of different diameter and different spans. These analyses are used to develop allowable support spacing for piping. The calculated allowable support spacing results in frequencies above the peak of the applicable floor response spectra. The simplified dynamic analysis method was applicable to all sizes of piping with design temperature less than 93.3 °C or to sizes of 2 inches and under for design temperatures above 93.3 °C. Piping of 2 inches or less nominal diameter was field routed using allowable support spans calculated by the simplified dynamic analysis method.

Per NPP Final Safety Analysis Report it is shown that »Postulated breaks in the reactor coolant loop, except for branch line connections, need not be considered for NPP. Subsequent to the General Design Criterion 4 final rule change (52 FR 41288, October, 1987), postulated breaks in the reactor coolant loop branch lines, pressurizer line, accumulator line and residual heat removal (RHR) line need not be considered.« The rupture of the 4 inch safety injection line into the reactor vessel down comer and the 6 inch safety injection line into the hot leg were considered. In addition, a one-square foot reactor vessel outlet nozzle break was considered for the control rod insertability evaluation of guide tube displacement. If a simultaneous seismic event, with the intensity of the SSE, is postulated to occur in conjunction with a loss of coolant accident, the combined loading must be considered. In NPP final safety analysis report it is shown that the stresses due to the SSE are combined in the most unfavorable manner with the blowdown stresses in order to obtain the largest principal stress and deflection. Thus, it is concluded that NPP meets the current criteria in General Design Criterion 4. In the NPP USAR, it is stated that mechanical components are classified in accordance with ANSI N18.2-1973. In the USAR, ANSI safety classes and ASME code classes are listed for systems in NPP.

It is noted that most of the Chemical and volume control system, Boron recycle system, Waste processing system and Spent fuel pool cooling system are classified as safety class 2 or 3 which would place them into seismic category 1. Some parts are non-nuclear-safety (NNS) but are limited to the parts that do not contain large amounts of radioactive waste.

All Category I structures are dynamically analyzed for SSE and OBE earthquake conditions using a modal analysis time history method. The Category I structures are mathematically modeled as assemblies of discrete mass point and stiffness elements. In the design, the reactor building, auxiliary building, intermediate building, fuel building and component cooling building are interconnected at floor and roof levels and are supported on a common foundation or island; the total complex of buildings is represented by a single seismic model. The diesel generator building and the essential service water pump house are modeled separately. In all the models, soil structure interaction is modeled using springs. The main island utilizes thirty-six such springs while the diesel generator building and pump house each has six springs. In all cases, the soil spring stiffness values are determined by estimating the soil strain under the predicted earthquake and therefore, are different for the SSE and the OBE. All the models are completely three dimensional, allowing six degrees of freedom (3 translational and 3 rotational) for mass points and stiffness elements. The time history earthquake is inputted independently in each of three orthogonal directions. The final seismic analysis of the three Category I structural models consists of a total of 18 time history analyses.

As can be seen from USAR, NPP complies with RG 1.29 and this compliance is accomplished through the application of ANSI N 18.2. Safety Class 1, 2 and 3 systems defined by ANSI N 18.2 are designated Seismic Category I and list of the related safety classification of equipment is provided.

The original NPP design complied with RG 1.84 and RG 1.85 through Revision 7, May, 1976 for Class 1 components. Thus, NPP complies with Regulatory Guides in the area of piping, component and component support of safety related systems and components as per ANS-N18.2 »Nuclear safety criteria for the design of stationary pressurized water reactor plants«.

Classification of systems and components by the ANS Safety Classes provides an adequate and proper determination of the applicable seismic design requirements.

The major Category I structures (except the diesel generator building and the essential service water pump house) are located on a common foundation and have continuous floor systems throughout the structure which minimizes any differential displacements. All floor slabs and major intermediate floor slabs are included in the seismic model; displacements and stresses that occur between floors are included in the design. When response spectrum methods are used to evaluate RCS primary components interconnected between floors, the procedures of the following paragraphs are used. The primary components of the RCS are supported at no more than two floor elevations. A dynamic response spectrum analysis is first made assuming no relative displacement between support points. The response spectra used in this analysis is the most severe floor response spectra. Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the integrated system analysis and in the detailed component analysis. The effect of the differential motion is to impose a rotation on the component from the building. This motion, then, being a free end displacement and being similar to thermal expansion loads, will cause stresses which have been evaluated with ASME Code methods used for stresses originating from restrained free end displacements. The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

NSSS Equipment

The Class I piping systems are analyzed to the rules of the ASME Code, Section III, NB 3650.

Reactor coolant loop piping

The reactor coolant loop (RCL) piping is analyzed using the time-history method on the coupled idling/loop system model by applying 6-component time-history accelerations (3 translational, and 3 rotational accelerations) at the base of the coupled system model. The coupled system model includes the Replacement SG model furnished by the manufacturer. The results of the RCL piping seismic analysis OBE and SSE include an amplification factor of 1.2 (an increase of 20%) to account for variations in the structural response parameters. In the coupled building/loop system model, effects of differential seismic movement of piping supports is automatically included in the analysis results. The analysis results are compared to the rules of the ASME Code, Section III, NB-3650.

Auxiliary Class 1 line analysis

The Auxiliary Class 1 Piping seismic analysis for OBE and SSE is performed using the Response Spectra modal analysis technique using the simultaneous occurrence of two earthquake components: one of two orthogonal horizontal and one vertical. Two 2D analyses are performed: first analysis for the response spectrum in the north-south direction and vertical direction; second analysis for the response spectrum in the east-west direction and the vertical direction. When response spectrum methods are used to evaluate piping systems supported at different elevations, the following procedures are used. The effect of differential seismic movement of piping supports is included in the piping analysis according to the rules of the ASME Boiler and Pressure Vessel Code, Section III, Paragraph NB 3653. According to ASME definitions, these displacements cause secondary stresses in the piping system. The following procedure is used to coordinate the effects of the various organizations involved in the seismic analysis, design and testing of Category I structures, components and equipment and to assure that appropriate seismic input data derived from seismic system and subsystem analyses are correctly specified to the manufacturer of Category I components and equipment and to construction of other Category I structures and systems:

1. NPP free-field ground response spectra for SSE and OBE generated on the basis of site geology and seismology surveys has developed artificial free-field time-history motions compatible with ground response spectra at the various damping values of interest
2. The seismic input data is distributed to the Krško NPP and all design participants who may require it for dynamic analysis or testing of Category I structures, systems or equipment
3. The free-field ground response spectra and time-history motions, together with all other required site and structure dependent parameters, as discussed in preceding paragraphs of this section, are utilized in performing dynamic analyses of the Category I structures and systems. In addition to the responses to seismic excitation required for design of the structures, response spectra has been developed at all points where they are required for dynamic analysis or testing of piping or equipment
4. Results of dynamic analysis are checked and reviewed by experienced specialists within Architect Engineer. Results are also verified by comparison with hand calculations and with outputs previously obtained for other projects to the extent that valid comparisons can be made.
5. Seismic responses and appropriate response spectra are transmitted to equipment suppliers by incorporation into the equipment specifications. Where required, additional information such as stiffness characteristics, effective masses and free vibration characteristics of supporting structure are also made available. The specifications also

require the supplier to describe, for prior approval by the procuring organization, the analytical methods or test procedures, including quality programs, which he proposes to use to qualify his product.

6. To ensure that the seismic design criteria are met, the above mentioned procedure is implemented for safety related mechanical equipment that falls within one of the many categories which have been analyzed as described and has been shown to be relatively rigid with all natural frequencies greater than 33 Hz:

Equivalent static acceleration factors for the horizontal and vertical directions are included in the equipment specification. The vendor must certify the adequacy of the equipment to meet the seismic requirements. When the floor response spectra are developed, the cognizant engineer responsible for the particular component, checks to ensure that the acceleration values are less than those given in the equipment specification.

Conclusions related to seismic regulations

As demonstrated in the SPSA, NPP was well designed for strong motion earthquakes. The seismic design criteria and the procedures used were state of the art at the time of the design and construction.

For NPP, single axis, single frequency tests were commonly conducted in accordance with IEEE 344-1975 and Regulatory Guide 1.100. The anchorage issues with equipment were addressed in the SPSA where a few equipment items were found to require anchorage upgrades which have been performed accordingly. Cable trays were found to be rugged in the SPSA evaluation.

2.1.2.4 Main operating provisions

Control room operator notification

In case of any seismic activity of sufficient intensity to activate the seismic instrumentation, the control room operator will be alerted by means of an alarm light and activation of the control room annunciation system. The operator will then obtain printed report and plots of the response spectrum in all three axes and associated Cumulative Absolute Velocity (CAV) values. These records numerically and graphically indicate any OBE exceedance against preset, allowable, acceleration amplitudes. System will be triggered (and all accelerographs record synchronously) by primary sensors located at down hole and free field. The system provides automatic on-line OBE analysis, printed report, and annunciation in the control room in case of exceeding of spectral accelerations and CAV for the designed OBE earthquake at free field and down hole and EPRI Check analyses (Response Spectrum & CAV Coincidence per RG 1.166/1997 – section 4) for free field and down hole accelergraphs. After OBE exceedence alarm annunciation appropriate post-earthquake action must be applied per RG 1.167. Section 3.7.4 of the USAR describes the seismic instrumentation program. For that purpose plant specific procedures exist to provide adequate instructions for plant personnel to respond on earthquake. When system is triggered operators will enter Alarm Response Procedure (ARP), followed by the Abnormal Operating Procedure (AOP) dealing with immediate and follow-up actions after seismic event. The administrative procedure is used for assessment of the plant status by verification that identified deficiencies/earthquake damage has been properly recorded and corrective actions performed as required to assure further plant operation.

2.1.2.5 Indirect effects of the earthquake taken into account

2.1.2.5.1 The failure of SSCs

In the development of fragilities for the NPP SPSA, the failure of SSC’s where specifically taken into account in evaluating equipment and structures. The structural response analysis used to develop in-structure spectra for NPP was conducted in a very conservative manner, and this excess conservatism more than compensated for any possible under-prediction of loads due to the method of combining earthquake components. In the reconciliation analysis of Class 1 piping, the analyses were conducted using the earthquake component combination criteria of RG 1.92 and no issues were uncovered relative to the original design basis combination of earthquake components.

Other important indirect effects of the earthquake on the SSCs have been analyzed such as:

- Short circuit on 6.3 kV safety bus:
    i. The faulted 6.3 kV bus will be isolated. The corresponding 400 V safety features switchgear and motor control centers will be lost; however, there are redundant valves and auxiliaries connected to the redundant switchgear buses and motor control centers for safe shutdown.
    ii. The AC source for the corresponding battery chargers will be lost, but the battery will assume the load of the inverter and other DC loads.
    iii. The faulted bus will be isolated by protective circuit breaker action so that minimal 400 V auxiliaries will be lost.
    iv. The 400 V safety features switchgear and control centers will be lost. The corresponding battery will be affected. Sufficient redundant auxiliaries will be fed from the remaining switchgear and motor control centers for safe plant shutdown.
- Diesel Generator Failure

    The consequence would be same as described above (Short Circuit on 6300 V safety bus). Sufficient redundant valves and auxiliaries would remain in service for safe shutdown fed from the remaining diesel generator.
- Batteries

    In case of loss of the control power source of 6.3 kV and 400 V buses associated with the battery would be lost; the remaining redundant safety features would be unaffected. The DC feed to inverter would be lost. If the loss of the battery occurred during the first 10 seconds after initiation of diesel engine starting, some instrument circuits would be lost. Capability of the protection system to actuate during the first 10 seconds would not be affected since two channels would still be available.

2.1.2.5.2 Loss of external power supply

Results of stability studies indicate that three-phase faults (with back-up clearing lines for stuck breakers) on the 400 kV systems do not impair the ability of the system to supply power to the Class 1E buses. The conditions studied include faults which result in outage of a single line, two lines or one line and the unit. The Krško NPP's switchyard bus faults, far-end line faults and the Krško generator characteristics are considered in the studies. The stability study obtained critical fault clearing times of less than 0.25 s and re-energization 0.3 s after clearing. Actual line protection will enable fault clearing within 0.1 to 0.15 s.

After consideration of the likelihood of losing offsite power, the redundancy of the emergency AC power system is the next most important contributor to reducing station blackout risk. With greater emergency power system redundancy, the potential for station blackout diminishes, as does the likelihood of core damage. The NPP has two standby power supplies available to power safe shutdown equipment. These standby power supplies are not used as alternative AC power sources. A safety design basis is the independent capability of these power supplies to achieve and maintain safe shutdown with off-site power unavailable.

Calculations show that the NPP emergency diesel generators (EDG) reliability is greater than 0.95. In conjunction with these results duration of four hours was determined as the time period for which

NPP demonstrates the ability to cope with a station blackout event and achieve a safe shutdown under station blackout conditions.

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

Access from three directions to the site is provided from the local road Krško to Brežice passing north of the site. An access railroad about 2 km long is constructed from Krško station to the site and connected to the Ljubljana - Zidani Most - Zagreb railroad network. The Sava river is not navigable in the vicinity of the site for the transport ships but can be used for transportation of smaller equipment and people with floating boats. The Zagreb Airport, located approximately 50 km south-east of the site, has a 5 km long paved runway and all necessary instrumental landing facilities. Commercial aircraft weighing approximately 50 tons, and a negligible number of private aircraft, weighing about 3 tons, are using the facilities. All commercial flights into and out of Zagreb Airport are controlled by the airport tower on regulated flight paths. All commercial, heavy and private, light aircraft traffic is controlled within a 150 km and 30 km radius of the airport, respectively. The Ljubljana Jože Pučnik Airport, located approximately 80 km north-west of the site, has a 3 km long paved runway. All flights into and out of Ljubljana Jože Pučnik Airport are controlled by the airport tower on regulated flight paths. Besides Ljubljana airport there is also smaller military airport Cerklje, located 4.3 km to the south from the plant which is also used for civil purposes.

The impacts of all three airports were reevaluated in the PSA report that was prepared in June 2010 and revised in April 2011. This PSA analysis uses the latest available flight data for all three airports, detailed PSA modeling and state-of-the-art standards. Results show that the CDF for the aircraft accident is 2E-7 /yr or lower.

Besides that the Krško NPP has also prepared an analysis regarding the impacts of aircraft crash on the plant. It shows that the plant is well designed and built, and with additional severe accident management equipment available onsite, well prepared even on such events. This report due to its sensitive nature is confidential and will remain as such.

2.1.2.5.4 Fire and explosion

The fire protection system (FP) is designed to provide fire protection from fire hazards. The FP cannot prevent a fire from occurring, but does provide the facilities for detecting and extinguishing fires in order to limit the damage caused by a single fire. In addition to the FP itself, there are many design features of the plant which would also contribute to confining and limiting a fire condition. The building structures are constructed of fire resistive concrete. The power plant is divided into several buildings that are separated from each other by three hour fire walls. These buildings are: reactor building, auxiliary building, control building, fuel handling building, intermediate building, diesel generator building, turbine building, and component cooling building. In addition, stair towers, in all but the reactor building, are enclosed with fire rated walls. The two EDGs are separated from each other by a two hour fire rated wall. The turbine lube oil reservoir and lube oil conditioning equipment are in a room which is separated from other areas of the turbine building by two hour fire rated construction. Extensive vertical runs of cables, ducts, and pipes are either enclosed in shafts with all shaft openings sealed with a noncombustible fire rated material, or all openings around cables, ducts, and pipes passing through major floors are sealed with a noncombustible fire rated material. Large oil filled transformers are located outdoors so that a fire would not damage the plant buildings. In addition fire barrier walls are located between the individual transformers and between the transformers and any air louvers in the walls of the turbine building. This limits a fire condition to only a single transformer without affecting the turbine building interior or an adjacent transformer. The plant FP is designed in accordance with the intent of the requirements for fire protection for nuclear plants established in the following:

- The National Fire Protection Association (NFPA) - National Fire Codes.
- International Guidelines for the Fire Protection of Nuclear Power Plants. (Published by the Swiss Pool for the Insurance of Atomic Risks on behalf of the National Nuclear Risks Insurance Pools and Associations.).

In addition to, and in compliance with the above listed standards and guidelines, all US manufactured fire protection equipment is approved for fire protection use by Underwriters' Laboratories Inc. or Factory Mutual Engineering Corp. The FP is designed such that the operation or failure of any portions of the FP does not produce an unsafe condition. Floor drains are provided in all areas of the plant which are protected by sprinkler systems. Floor drains are also provided in all areas of the plant where fire protection piping is located in order to remove water discharged from any break in the FP or water discharged from portable fire hose streams in an actual fire. Other plant equipment is located on raised concrete pads to reduce the possibility of damage due to flooding. The routing of the FP water piping is in such a manner as to minimize exposure to Safety Class equipment. Each charcoal filter plenum is protected by a separate manually operated water spray system. Each main reactor coolant pump is protected by a separate independent manually operated water spray system. All manually operated fire extinguishing systems require the operation of two devices (cover and switch, or cover and valve) to discharge the systems. FP piping within the seismically qualified charcoal filters and near the reactor coolant pumps are supported to withstand a seismic event. All FP pipes penetrating the reactor building containment (supply pipes for certain charcoal filter water spray systems, for the main reactor coolant pump water spray systems, and for the standpipe hose stations within the containment) contain Safety Class isolation valves which are normally shut or which would automatically close in the event of a nuclear emergency. A failure mode and effect analysis of the FP has been performed showing that failure of any portion of the FP system would cause »Fire Alarm« and »Trouble Alarm« annunciated at the main control room (MCR) followed by appropriate plant procedure to deal with individual plant condition.

2.1.3 Compliance of the installation with its current licensing basis

2.1.3.1 Operator's general organization / process to ensure compliance

General description of plant organization

Krško NPP is organized for the operations phase as defined by USAR chapter 13 and 17. NPP has overall responsibility for design, engineering, construction, licensing, operation, fuel management, procurement and quality assurance. These responsibilities and tasks are performed within the following divisions: Technical Division which is responsible for operating, maintenance and technical services; Engineering Services Division responsible for design, engineering, configuration management, licensing, procurement engineering, project management and safety assessments; Quality and Nuclear Oversight Division, Administrative Division, Purchasing Division and Financial Division.

Directors of the above divisions report to the Management Board. Beside these divisions, managers of three departments report also directly to the Management Board; Training, Security and Information Systems.

The President of the Management Board is the head of the company who has direct line authority over all nuclear related operations and is currently designated as the »Engineer in Charge« as defined by ANSI N18.1 (1971). In absence of the President of the Management Board his function is performed by other Member of the Management Board. In general, the Management Board is composed of two members: the President of the Management Board and the Member of the Management Board.

The overall management of NPP activities is provided by the Management Board who reports to the Supervisory Board. The Supervisory Board, established by owners GEN energija d.o.o. and Hrvatska Elektroprivreda, functions to review and approve NPP budget and policy, major improvements and modifications.

The organization during the operations phase of the NPP is presented in such a way, that all personnel are performing quality-affecting activities onsite. The Quality assurance program includes all planned and systematic actions taken by Krško NPP, including the contractors and consultants, which provide adequate confidence that structure, systems and components shall perform satisfactorily in service. The documented program which consists of the Quality assurance plan and applicable Administrative, Technical Operations, Engineering Services and Quality Assurance procedures, is mandatory for all activities affecting the safety related functions of the nuclear power plant structures, systems and equipment, but may be applied also to non-safety-related items as deemed appropriate by the plant management.

The QA plan is addressing each of the criteria of Appendix B to CFR, Part 50.

The QA manual is the top level quality document for operational phase activities. The requirements identified by the QA plan shall be implemented according to management directives, programs, plans, procedures or instructions, grouped in plant level manuals, division level manuals and department level manuals and programs: Plant Management Manual, Technical Operation Manual, Engineering Services Manual, Quality Systems Manual, Purchasing Manual, Financial Manual, Information System Manual, Security Manual, Training and Indoctrination Manual.

Design control process

Design control process has been established to assure that applicable regulatory requirements, codes and standards and the design bases are correctly translated into design documents, such as specifications, drawings, procedures and instructions. QA requirements for design activities in the operational phase are established in the pertaining QA plan, Chapter III. QA Implementing Procedures, Administrative Procedures (ADP) and Engineering Services Procedures (ESP) for controlling design activities for the operational phase have been developed. Directives for design activities are documented in the Design Control Program which is one of the Engineering Services Manual programs. The directives include definitions of: design interfaces, documents and records control, quality levels, acceptance standards and record requirements, design activities audits and corrective actions performance, training and experience transmission requirements.

Design control shall be done for those projects comparable in nature and to the extent of the original construction. Design activities: Design Change Initiation, Design Inputs, Safety Evaluation, Design Process, Design Verification, Modification Implementation in the operational phase of Krško NPP are planned, implemented and controlled in accordance with ADP and ESP Design Control Procedures. Engineering support division (ESD) is responsible for initiating actions on requested design changes: establish a method for the evaluation by the Krško Operating Committee, provide for review and approval of proposed changes. ESD assigns personnel responsible for coordination, preparation and implementation of the modifications which includes identification, documentation, review and approval of applicable design inputs.

Design Input Requirements shall be in accordance with ANSI N45.2.11-74. QA requirements for the design of NPP changes from the specified design inputs are identified, approved and controlled. A safety evaluation which considers the effect of the design change is provided by ESD. The safety evaluation is reviewed by the Krško Operating Committee and prepared as a basis for the review by Krško Safety Committee, to verify that the modification is properly evaluated per 10CFR50.59.

Procurement and document control

Procurement control is established through the process defined by written procedures which define the requirements, responsibilities and the sequence of actions to be accomplished in the preparation, review, approval and changes to procurement documents.

Technical operations department is responsible for identification of materials, parts, equipment and services needed for plant maintenance and operation, identifying their safety classification and planning and scheduling of procurement on the bases of needs. They shall provide also detailed description of the scope of supply, references to drawings and other design documents.

Engineering services division is responsible for preparation and control of technical and purchasing documentation, and incorporation of design, technical and regulatory requirements. They are responsible also for identifying supplier's documentation to be generated, delivered or maintained for review, approval and information and for providing review and approval of procurement documents for safety and seismic related items.

Quality assurance systems division is responsible for defining QA program requirements to be included into procurement documents and providing support for definition of requirements for supplier's records control and submittal. They are also responsible for verification that suppliers are on the Approved Suppliers' List, that all quality requirements, including 10CFR21 (if applicable), are properly included into procurement documents. They perform review of suppliers QA programs and audits implementation.

Document Control in Krško NPP is performed in accordance with NPP Document Control Program and the pertaining procedures. Information System provides by the Document Control Module on-line document data access for plant personnel with direct Document-to-Equipment and Procedure Index Cross-Reference database. Controlled documents are secured and safely filed in the Central Vault.

Regulatory requirements

Each design or other change performed on the plant is subject of the review process involving 10CFR50.59 process or legislative review per JV9 regulation »Rules on operational safety of radiation or nuclear facilities«, which predicts different categories of changes and their way of resolution and regulatory approval. In case that change is significant or involves Technical Specifications change, regulatory approval is needed before implementing the change (i.e. category 3 per article 83 of the nuclear law). Lower categories require agreement by regulator before particular change implementation (i.e. category 2 per article 83. of nuclear law) and information about the change if it has been categorized into the category 1.

The NPP was originally designed and licensed based on NRC regulation which is used in several areas where domestic legislative requirements are not directly regulating or required. The NPP is periodically updates a document called »Regulatory Conformance Program - Compliance Review« which represents the overview of all US NRC regulatory requirements and NPP compliance with them. This document is also a part of the PSR process. Currently, revision 3 is in place.

Conclusion

NPP organization, established processes and operational practices are defined in top level plant documents defining proper design control and potential controlled and regulated changes on the plant. Thus, the compliance with design bases standards and regulatory requirements is achieved through the whole plant lifetime.

Likewise the plant is performing regular test and maintenance of the equipment in the framework of preventive and corrective programs. In the case of seismic event the equipment is checked in accordance with the procedure ADP-1.2.301 »NEK response to an earthquake«.

2.1.3.2 Operator's organization for mobile equipment and supplies

Krško NPP organization in case of emergency is defined and described in chapter 5 of this report. Several types of procedures exist: abnormal operating procedures (AOP), emergency operating procedures (EOP), and the severe accident management guidelines (SAMG – see details in chapter 6.2.1.2.2). The main required equipment is installed and available on site including mobile equipment required to cope with all design and beyond design bases accidents.

The organizational aspect for mobile equipment includes availability of the equipment, appropriate supply of needed supplies for their operations, periodic maintenance and surveillance testing and training for personnel.

The maintenance of equipment procured to support the strategies and guidance during severe and abnormal plant conditions is performed within plant organization and configuration control. All equipment is maintained in accordance with manufacturer instruction manuals and in house practice for similar existing equipment in the plant. Requirements are specified in plant maintenance department procedures. Frequency of maintenance is defined in preventive maintenance programs.

All portable diesel engines are in standby and are ready for use in case they would be required. The testing and periodic maintenance process is established through plant programs and procedures (once per three months test and routine inspection and standard service are performed once per year). The complete list of SAME mobile equipment is provided in Appendix I of this report. The equipment consists of portable electrical (125 V DC, 118 V AC) generators, compressors, mobile diesel generators, fire protection pumps, submersible pumps and transformers. All other equipment (valves, flanges, connection cable,) is part of standard preventive procedures. Testing is performed by verifying that each diesel or gasoline engine is started and capable to perform its design function. The corrective action process established on the plant is used as a process when the equipment fails to adequately perform its test or function.

The controls for assuring that the equipment is available when needed are established through regular testing and maintenance. The inventory requirements established for the equipment are assured with the general design and installation and system operating procedures which require necessary inventory for operation for 3 days without any external support. Besides inventory requirements, all equipment spare parts recommended from manufacturers are on stock labeling of equipment is performed according to plant procedures.

Potential plant configuration changes on mobile equipment are part of the plant equipment and configuration control and will be applied to the new installed equipment on the same way as this is established for other plant systems, structures and components.

As a part of plant changes, procedure modifications or any new procedure follow the standard process for validation and approval. Training materials are developed to support the activities related to the strategies including simulator training of operation and support personnel.

All new jobs and tasks from the mitigating strategies performed by the personnel are analyzed and added to the Job-and-Tasks-Analysis data base. According to the new tasks, appropriate training materials and simulator scenarios are developed for initial training, qualification and requalification process for operators, auxiliary operators and other support personnel.

2.1.3.3 Deviations and rework

The NPP developed a PSHA analyses in 1994. The resulting hazard determined within this study was used as an input for SPSA calculations. The results were valid for free field and direct comparison with design spectra were recalculated due to the known effect of soil-structure interaction. In a probabilistic response analysis, the characteristics of the free-field ground motion are defined by the shape of the median uniform hazard spectrum corresponding to a return period of interest. For the NPP, the uniform hazard spectrum shape corresponding to a 10000 year return period was used. This uniform hazard spectrum was developed from a hazard analysis of the site. The seismic level of the analyses was twice the SSE level, since the SSE level for NPP is 0.3 g and median uniform hazard spectrum for the probabilistic analyses were anchored to PGA of 0.6 g.

The probabilistic analysis was carried out at an earthquake level equal to twice the design earthquake level. The response quantities of interest recovered from the multiple earthquake simulations included peak accelerations, maximum member forces, and floor acceleration time histories. These quantities were needed for fragility development. For the three structures analyzed, the median centered spectral peaks occur at lower frequencies when compared to the USAR results. The frequency shift can be easily explained as follows. Since the probabilistic analysis was performed at a higher earthquake level than the USAR analysis, the median soil stiffness properties for the probabilistic case are lower than those for the USAR case. In average, the ratio between the soil shear wave velocities is of the order of 0.6. Since the responses of these structures are mainly controlled by the soil, a shift on the dominant frequency of the responses of about 0.6 is also expected.

In the years 2002 to 2004 a revised PSHA was performed for Krško NPP site. The resulting frequency spectra of accelerations were calculated from the history of earthquakes in the region and from the activity of faults near the location of NPP. Different data was evaluated by independent groups of experts to gain a common probability and intensity of probable maximum acceleration at the location of the NPP. The aim of additional studies was to evaluate the vicinity of the NPP with total of 45 km seismic profiles.

As a part of the SPSA investigations for the NPP, IPEEE analysis for the seismic part was performed in the nineties. An IPE was performed for the NPP including the IPEEE. An important task in the IPE and the IPEEE was a detailed walkdown of the plant to identify seismic vulnerabilities and any plant specific features which are important for the derivation of seismic fragilities. Vulnerabilities associated with containment overpressure capacity or sources of containment bypass were also assessed via a walkdown.

Within the scope of the containment walkdown performed in 1992 during refueling outage, all safety related components, piping; instruments, tubing and cabling and all mechanical penetrations were inspected.

In general the conclusion was that NPP had been well designed and constructed for seismic events. One observation was that there are an excess amount of snubbers used. This is the result of following very conservative criteria in existence at the time of the design.

There were no serious seismic issues observed in containment.

In 1994 the walkdown outside containment was performed, covering all components outside of containment, which were identified in the IPE as essential components for accident mitigation and safe shutdown of the plant.

The safe shutdown systems and components were identified through NPP systems analysis for the IPE, and hold for both internal and external initiating events. The Safe Shutdown Equipment List (SSEL) was the basis for a walkdown focused on passive systems or components which have structural failure as their only failure mode. The review included piping, cable trays, HVAC ducts, II/I issues, steel structures and large tanks (RWST, condensate storage tank, buried diesel fuel oil tanks).

Outside containment walkdown observations were identified for different groups of equipment, including:

- Support and Anchorage
- Building Differential Motion
- Cabinet Impact Issues
- System Interactions

– Evaluation on proposed fragility evaluation

For all identified observations, the NPP performed appropriate corrective actions or design changes and resolved all deviations.

As part of the preliminary activity to update seismic fragilities, a special walkdown was conducted of the containment during the outage in May, 2003 to assess the potential for seismic failure of instrument tubing connected to the primary system and to observe the upgrades made to the flux monitoring cart. In addition, a walkdown was conducted of the 110 kV and 400 kV switchyards and the 110 kV power source in the Brestanica power station. Some additional walkdowns were conducted of components in the control, auxiliary and intermediate buildings to examine upgrades made to component anchorage and supports since the original Probabilistic Seismic Response Analysis for Krško NPP (PSRA, 1995). Also, a 0.6 kV portable diesel generator at the site was examined for purposes of gathering information for development of fragility for this machine. Revised fragilities were developed for the 400 kV offsite power and new fragilities were developed for the 110 kV offsite power and the 0.6 kV diesel generator. It was also noted in the walkdown that additional upgrading was needed for some components. Additional upgrading was needed for the CCW surge tank, the protective relay racks and the safety related motor control center MCCD 221.

In December, 2003 a more detailed walkdown was conducted outside of containment of all components noted in the 1994 walkdown report to require upgrading or additional assessment or to assess new equipment added or replaced since the 1995 PSRA.

Results of the walkdown can be summarized in the following areas:

Control room ceiling:

A new control room ceiling had been added since the original PSRA. The control room ceiling was originally noted to not have been designed for seismic events and that it would likely collapse in the strong motion events postulated at Krško. The new ceiling was installed in accordance with requirements of the Uniform Building Code as applied to Seismic Zone 4 in California. The new ceiling would remain intact in case of SSE event.

Refueling water storage tank:

The RWST is located on a thick concrete slab in the yard next to the auxiliary building and is well anchored to the base mat together with a reactor water makeup tank. Both tanks are enclosed in a concrete wall. Upon failure of either tank, the water would be retained within the concrete shield wall providing that the piping penetrations into the base slab are sealed. The RWST dominant failure mode is sliding which will result in failure of the connecting piping. If the RWST piping fails, the tank will drain down to a lover level but the essential function of the RWST will not be lost. The important feature is to have penetrations sealed.

Component cooling water pumps:

The CCW pumps are typically a low pressure high volume pump with large piping connected. The piping reactions on the pumps were examined as a seismic issue. It was observed, that the piping into and out of the pumps was anchored to the floor, thus there are effectively no seismic induced piping reactions on the pump nozzles and the existing pump anchorage.

Several observations in past walkdowns required verification and demonstration, that the supporting anchorage and installation is adequate and represent higher capacity than target 2.75 g median PGA screening capacity. In the 1995 PSRA the screened out level was above 2.0 g. In many cases the calculated fragility was much higher and the accumulated conservatism of only crediting the SSCs with 2.0 g median capacity tended to overestimate the CDF. The screening level has been raised to 2.75 g which will result in a lower probability of failure when convolved with the new hazard than the previous 2.0 g fragility when convolved with the 1994 hazard.

The new seismic hazard screening level rose from 2.0 g to the higher level of 2.75 g median capacity associates components with fragility rate of less than 2E-7 /year which is less than 1% of the expected final CDF from seismic events. This <1% target is compatible with guidance in the ANS Standard and in the EPRI SPRA Applications Guide (EPRI, 2004).

Examples of such observations and verifications were CCW surge tank, high pressure safety injection (HPSI) pump air handler, Diesel generator building fan, ducting and power supply, diesel generator temperature and pressure switches, diesel generator fuel and lube oil filters.

Penetration Areas:

There were several potential issues noted in the 1994 walkdown report (1994) regarding the effect of differential motion between buildings on containment penetrations. The containment overpressure evaluation (1993) determined that the penetrations were much stronger than the piping, thus the differential building motion between the containment/shield building and the auxiliary building could only affect piping integrity outside of containment or the piping supports. The issues noted were piping supports in the auxiliary building close to the containment penetration and piping supports from both the shield building and auxiliary building that restricted piping movement in the same direction. The 1995 PSRA was based on the assumption that the issues noted in the 1994 walkdown report had been resolved by modifications or analysis. These potentially issues were revisited and found to be adequately resolved.

Other minor observations on different groups of equipment were demonstrated to be adequately resolved either by performed modifications/corrective actions or by the original design:

Based on several reviews and walkdowns reports, NPP systematically performed several modifications and small corrective actions that support NPP earthquake resistance.

The most important modifications already performed are the following:

- Ceiling replacement in MCR
- Switchgear anchoring improvement
- CCW surge tank anchoring improvement
- Follow up for CCW surge tank anchor improvement
- Improvement of DG cabinets
- Lighting distribution panel bolting improvement
- Waste processing gas system compressor unit improvement
- Polar crane seismic improvement
- Seismic protection of polar crane
- Installation of seismic stabilizer
- Improvement of EDG fuel tank supports
- Seismic reinforcement of IA tanks
- Improvement of nitrogen bottles attachment

The first PSR represented a significant review process where seismic issues were identified, evaluated and new actions setup for plant seismic improvements. One of the most important contributors will be installation of new third emergency diesel generator, which will be completed in 2012.

2.1.3.4 Specific compliance check already initiated by the operator

NPP performed the analysis in accordance with NEI 06-12 B.5.b »Phase 2&3 Submittal Guideline«. The result was the proposal of corrective actions regarding:

- Strategies to decrease the loss of reactor coolant,

- Strategies to improve residual heat removal by the steam generators,
- Divers means of water supplies to the SFP,
- Strategies to flood the containment with alternative equipment,
- Strategies to supply water to different sources (e.g., CST, RWST, water treatment system, pretreatment water system, demineralized water system).

Detailed description of implemented actions was delivered to the SNSA.

2.2 Evaluation of margins

2.2.1 Range of earthquake leading to severe fuel consequences

2.2.1.1 Weak points and cliff edge effects

Approach

The approach to evaluating seismic margins at NPP consists, in general, of the following main steps:

- From the USAR and available safety studies, identify the »success paths« for a range of seismic events. A »success path« is defined as a minimum set of functions required for avoiding reactor core damage state following an earthquake. Each success path identified is specified in terms of required critical safety functions.
- Map each critical safety function in every success path to the specific plant's SSCs.
- For each relevant SSC, determine from the existing plant specific safety / risk studies its »margin« for an earthquake, or a »seismic margin«. A seismic margin can be expressed as the highest earthquake for which considered SSC has low probability of failure, as estimated with high confidence. For example, seismic margin can be determined on the basis of so-called High Confidence of Low Probability of Failure (HCLPF) value, considering also the median seismic capacity, due to the range of uncertainty involved in seismic fragility assessment.
    i. Generally, seismic capacity for a system is defined on the basis of controlling failure mode (i.e. the weakest point).
- Based on the individual seismic margins for relevant SSCs, determine the representative seismic margin for each critical safety function. In doing this, use the following general principles:
    i. When the function is provided by multiple SSCs where each SSCs is necessary for proper function performance, the representative margin is the lowest margin of SSCs involved.
    ii. When the function can be provided by several SSCs where each SSCs is sufficient for proper function performance, the representative margin is the highest margin of SSCs involved.
- Evaluate the availability of success paths following a postulated seismic event, with increasing severity. Start with the lowest seismic events and gradually increase the severity, in terms of PGA. Critical functions for which the corresponding seismic margin has been exceeded are assumed unavailable. In this manner, all relevant cliff edges are identified. A success path becomes disabled when first of the required critical functions becomes unavailable. With increasing seismic severity, a number of success paths would be smaller and smaller. The point (seismic level) at which the last success path is disabled can be considered a »seismic margin« for the whole plant.

Similar approach can be taken for evaluating the SFP and Containment performance.

Brief history overview

The OBE and SSE levels were defined in the USAR based on the existing geological and seismological information at the time and on the existing requirements for the plant in the US. Input spectra and response spectra, with seismic design requirements on the plant SSCs were set accordingly.

In early nineties, a detailed IPE was undertaken, including internal and external events and hazards. As a part of it, a PSHA was performed for the NPP site, collecting ant evaluating the information relevant for seismic hazard which was available at the time. This assessment provided detailed characterization of seismic hazard at NPP site (in the form of hazard curves and hazard spectra), which was then considered in the PSRA and SPSA.

In the PSRA, which was conducted under the IPEEE, a detailed assessment was performed of floor response spectra, based on the input hazard spectra anchored to 2 × SSE. These spectra formed the basis for the analyses of seismic fragilities and capacities, which were then used for the quantification of seismic risk in the SPSA.

SPSA for NPP, done under the IPEEE in mid-nineties, together with mentioned seismic hazard analyses and evaluations of seismic fragilities of equipment and seismic capacities of structures represented, at the time, one of the most comprehensive seismic risk studies for a NPP.

In the early 2000s, the PSHA for NPP Krško site was revisited as a part of the first PSR. The revised PSHA reflected the results of latest, at the time, seismological and geological researches and studies, resulting in the revised seismotectonic model for Krško basin.

Revision of the seismic hazard characterization required revision to seismic risk analysis. For this purpose, scaling was performed of the existing probabilistic response spectra in order to reflect the updated seismic hazard data. This resulted in the updated (scaled) floor response spectra which formed the basis for the update to the assessment of seismic fragilities and capacities.

The above described activities led to the comprehensive upgrade to the SPSA for Krško NPP (including PSA Level 1 and Level 2), which was performed, roughly, a decade later than the initial seismic PSA study. This study provided the upgraded logic and probabilistic model for evaluation of the risk from the seismically induced accident sequences.

Also, it needs to be pointed out that NPP was subject to large modernization program in 2000, which included power up-rate and replacement of SGs. As a part of it, the stress analyses to NSSS were revisited.

Identification of success paths

Success paths for NPP following a seismic event can be summarized as follows.

In the case that seismic event induces a large brake LOCA, the required critical function for success (i.e. avoidance of core damage) would be low pressure safety injection (LPSI) and recirculation.

If induced initiator is medium brake LOCA, the required critical functions for the success would be reactor scram and HPSI / recirculation. (It should be noted that the latter also involves LPSI function - LPSI Pumps to provide suction for HPSI pumps.)

In the case that induced initiator is small brake LOCA, the required critical functions for the success would be reactor ccram, HPSI / recirculation and secondary heat sink (AFW). (There is, also, an additional success path with RCS depressurization, AFW and LPSI injection / recirculation.)

If none of the LOCA categories is induced, but there is a total loss of ESW, the required critical functions would be reactor scram, secondary heat sink (by means of AFW TDP) and RCS inventory / RCP seal integrity.

If none of the above categories is induced, but there is a Steam Line Break (SLB), the required critical functions would be reactor scram, HPSI / recirculation and secondary heat sink (by means of motro driven AFW pumps). (Note: There are also some other success paths involving successful main steam isolation and avoiding the need for high pressure recirculation, which are, conservatively, not considered.)

If none of the above is induced, but there is a seismically induced failure to insert control rods (i.e. seismically induced Anticipated Transient Without Scram, ATWS), the required critical functions would be secondary heat sink (AFW), pressurizer relief and long term shutdown (by means of boric acid transfer pumps and CVCS charging pumps).

If none of the above initiators is induced, the Loss of offsite power (LOOP) should be considered. In such a case, the required critical functions would be onsite power (by means of EDGs), secondary heat sink (AFW) and RCP seal injection (by means of CVCS charging pumps or PDP). (For the last one, there is an additional success path: CCW to RCP thermal barriers).

Finally, it is pointed out that all the above success paths additionally require success of the following two functions:

- Integrity of large structures (e.g. buildings);
- Integrity of large primary system components (e.g. reactor vessel, SGs, etc.).

Failure of any of these two would lead to beyond design basis conditions for which no success path can be considered.

Evaluation of plant level seismic margin

According to the described approach, the plant level seismic margin is assessed by evaluating the availability of success paths following a postulated seismic event, with increasing severity. The evaluation is based on consideration of controlling seismic failure modes for the relevant SSCs, including necessary support systems.

The evaluation of plant level seismic margin starts with the lowest seismic events and continues with gradual increasing of the seismic severity, in terms of PGA.

Based on the plant specific seismic fragility analyses, the expected plant response would be as follows from core damage standpoint.

Earthquakes in the range below the OBE (PGA < 0.15 g)

At earthquake levels approaching OBE value, the 400 kV switchyard (HCLPF = 0.093 g) would fail with probability close to 10%. However, a failure of 110 kV is considered low probability event (HCLPF = 0.15 g, Median Capacity = 0.29 g; therefore, failure probability of 1% at most) at this seismicity level. Therefore, a complete LOOP is not considered likely. (Failure probability is considered bounded at 0.1 × 0.01 = 1E-03 per event.) Failures of any safety related SSC are considered unlikely (as the upper end of the considered interval is well below the HCLPF values for the relevant SSCs).

The expected plant response can be bounded by reactor trip due to interruption of normal power supply, followed by a bus transfer to 110 kV and normal response of plant systems. At lower levels, 400 kV would not be interrupted and plant can be expected to respond with normal reactor trip or normal administrative shutdown.

Earthquakes in the range between the OBE and SSE (0.15 g < PGA < 0.30 g).

Toward the upper end of this seismic interval, a LOOP can be expected. (Median capacity of 110 kV offsite power is around 0.30 g and, hence, failure probability would approach 50%.)

Failures of safety related SSC are considered unlikely (as the upper end of the considered interval is well below the HCLPF values for the relevant SSCs).

Therefore, the expected sequence in the range of 0.15 g to 0.30 g can be bounded by a LOOP without additional failures of safety related SSCs. At lower part of the interval, the expected sequence is reactor trip with, at worst, 110 kV offsite power available.

Earthquakes in the range between the SSE and 0.45 g (0.30 g < PGA < 0.45 g).

The expected sequence in this range is considered to be a LOOP without additional failures of safety related SSCs. (Median capacity of 110 kV offsite power is around 0.30 g. Failure probabilities of safety related SSCs are below 1%.

Earthquakes in the range of 0.45 g < PGA < 0.60 g.

None of the discussed success paths is still considered to be affected in this seismic range. The expected sequence in the range is still a LOOP with possible, although not likely, additional failure of condensate storage tanks (CST) and / or RWST (probability < 6%). Success path for LOOP (in the absence of any other initiator) would apply. Suction to the AFW pumps would be provided from the CST (if not failed) or from the ESW. The RCP seal injection would be provided by the CVCS pumps taking suction from the volume control tank (VCT). Power would be provided by the EDGs. Regarding the RWST the following needs to be noted: the initiators requiring the injection from this tank are low probability events in this seismic range. The RWST availability is required for the LOCAs and SLB. The corresponding SSCs (i.e. the primary or secondary piping) are not expected to fail, with high confidence, at ground accelerations lower than 1 g. Therefore, those are very unlikely sequences in this range. For the remaining success paths (i.e. those related to induced initiators other than LOCA / SLB), the RWST is a part of an alternative success path for the RCP seal injection (in the case of failed CVCS pumps suction from the VCT). However, considering the seismic capacity of CVCS Pumps and VCT (above 1 g), this function is not of a concern in the considered seismic range. Therefore, even if the RWST failure occurs (which is not likely), it would not affect the success paths in this range.

Earthquakes in the range of 0.60 g < PGA < 0.75 g.

With earthquakes in this range, structural failure of CST and / or RWST is a credible consequence. (At the upper end, failure probability is 17% for the CST.)

Failure of RWST is not considered a concern, as far as induced LOCAs /SLB are considered, due to their high HCLPF values. The RWST is also alternative source for CVCS pumps to perform RCP seal injection. This function (suction to CVCS pumps) can, however, be achieved by alternative means described in the EOP ECA-0.0 Appendices. This range of earthquakes includes the HCLPF levels for the reactor scram failure (based on the impact on fuel assembly geometry) and Loss of ESW (due to ESW pump house seismic failure modes; HCLPF value for ESW pump house is 0.68 g). However, considering the median capacities as high as 1.62 g for fuel assembly geometry and 2.54 g for ESW pump house, these failure modes are not considered likely in this interval. (Failure probabilities: < 2.8% for fuel assembly geometry and < 1.6% for ESW pump house.)

Failure of EDGs (due to electrical periphery) is also not considered likely in this interval (Median Capacity is 1.45 g. Failure probability < 6%).

Assuming the failure of CST, alternate suction would be established for the AFW pumps by re-alignment to ESW.

Assuming the additional failure of EDGs, although it is not likely in this range (failure probability < 6%), alternative means would be required to support the success paths considered, as EDG availability is necessary for all seven success paths specified earlier. Alternative means, as described in the EOP ECA-0.0 Appendices, will be needed to ensure secondary heat sink and RCP seals integrity.

For each of these functions, it is considered that at least one option would exist by means of alternative equipment which is located at the free field with housing to be provided at the nearest future by metal construction designed for at least 2 × SSE. For CVCS PDP, this is a mobile 400 V diesel generator. For secondary heat sink, this are portable fire pumps. (Also, it is pointed that two mobile air compressors are available at the site with seismic and load capacity higher than existing IA compressors. They are located at the free field together with other mobile equipment.) Such equipment is considered to be seismically robust and having considerable seismic capacity. As an example, the Krško NPP SPRA documentation provides for the existing small portable 400 V diesel generator (0.6 kVA) the HCLPF value of, approximately, 1 g. Therefore, from the seismic fragility point of view, this equipment is considered to be able to support the success path for LOOP at seismic levels which may imply failure of EDGs.

It needs to be pointed that implementation of alternative means requires locally performed manual actions in relatively short time. The SPSA indicates that critical time would be the time for establishing the RCP seal injection from the PDP before the seals degrade. However, with instructions provided in the emergency procedures (specifically, ECA-0.0), it is considered that the necessary actions can be performed successfully. It is, therefore, considered that the success path for LOOP can be supported at seismic levels up to, at least, 0.80 g, at which the EDG operation may not be credited.

Earthquakes in the range of 0.75 g < PGA < 1.0 g.

At this interval, seismic failure of EDGs is considered likely (at 0.85 g, probability ≤ 10%). This means that alternative means will be needed to ensure secondary heat sink and the RCP seals integrity. Those were discussed above.

Additionally, at the upper part of the interval, failure of control rods insertion cannot be excluded. (At the upper end, probability of failure for the fuel assembly geometry is ≤ 10%.) The loss of ESW pump house is still relatively unlikely. (Failure probability at the upper end is < 5%.) Loss of ESW, however, would not have the additional impact, as EDGs would, likely, be also lost at this level. Therefore, critical functions would be those associated with seismic ATWS sequence. Long term shutdown (sub-criticality) can still be achieved, considering the high seismic capacity of boric acid transfer pumps / tank and CVCS PDP. RCP seal integrity would also be ensured by CVCS PDP with suction aligned to alternative means (ECA-0.0 Appendices) However, the critical function is ensuring the secondary heat sink. It is questionable whether, in the ATWS sequence, the alternative suction to AFW pumps can be ensured in time.

Also, it needs to be pointed that in the upper part of this interval the liquefaction cannot be excluded (HCLPF = 0.8 g; probability at the upper end ≤ 14%.).

To summarize:

- At the lower end, the expected sequence is seismic SBO with structural failure of CST and RWST;
- Toward the upper part of the interval, the expected sequence is seismic ATWS with SBO conditions (which encompass the loss of ESW); the liquefaction, also cannot be excluded.

Earthquakes in the range of PGA > 1.0 g.

At seismic levels of, approximately, 1 g, a number of SSCs are expected to fail, including CST, RWST, EDGs and ESW. Certain degradation of fuel assemblies' geometry in the core is also expected, which can prevent the control rods to drop in the core, causing the reactor scram failure. At seismic levels exceeding 1 g, failures of other safety systems, as well as larger structures, also cannot be excluded. For seismic levels of 1 g and higher, no success paths for preserving reactor core are considered.

Conclusion regarding the seismic core damage margin – cliff edge effect

Based on the seismic margin evaluation, taking into account the alternative means described in the EOPs and SAMGs (see details in chapter 6.2.1.2.2), it is considered that seismic levels at which core damage would be likely are at PGA range of 0.8 g or higher. At these seismic levels, the critical induced sequence is seismic ATWS with SBO conditions. Seismic ATWS could, at seismic events of such a severity, be caused by a failure of control rods insertion due to degradation of fuel assemblies' geometry. Although the long term shutdown (sub-criticality) can still be achieved (boric acid transfer system), the critical function is ensuring the secondary heat sink in time. Following the seismic failure of CST, together with conditions of induced SBO and / or loss of ESW, the secondary heat sink would have to be provided by alternative means specified in the EOP ECA-0.0 Appendices. It is questionable, however, whether this can be implemented in time, considering the ATWS condition.

Also, the liquefaction cannot be excluded which would potentially fail buried structures and / or equipment.

Seismic capacities of structures related to primary or secondary pipe breaks (i.e. LOCAs or SLBs) are fairly above these levels.

At the end, it needs to be pointed out that seismic events with PGA higher than 0.8 g were estimated to be very rare events at the Krško NPP site. Based on the revised PSHA and SPSA, the return period for such an event is considered to be larger than 50000 years.

2.2.1.2 Measures which can be envisaged to increase robustness of the installation

A number of measures have already been implemented at NPP plant in order to increase the robustness with respect to seismic events. They include:

- Alternative means to provide suction to AFW pumps or to provide water to SGs directly;
- Alternative means for power supply to CVCS PDP in order to preserve RCS inventory and the integrity of RCP seals in induced SBO or Loss of ESW / CCW conditions;
- Alternative means for power supply to selected MOVs, as necessary for the implementation of alternative methods;
- Alternative means for providing water from the external sources to containment;
- Procedures for local operation of AFW TDP and for local depressurization by means of SG PORVs, both without need of DC or instrument power;
- Alternative means for makeup of SFP inventory.

2.2.2 Range of earthquake leading to loss of containment integrity and SFP integrity

Approach

For evaluating seismic margin for the containment integrity, an analogous approach was taken to the one described in section 2.2.1.1, which was applied to reactor core damage margin.

It consists, in general, from the following main steps:

- Identify, based on the USAR and available safety studies, the »success paths« for the containment integrity following a range of seismic events. A »success path« is in this case defined as a minimum set of functions required for avoiding radioactivity releases into the environment following an earthquake. As for the core damage, each success path identified is specified in terms of required critical safety functions.

  i. Success paths are defined for early releases and for late releases separately.

- Map each critical safety function in every success path to the specific plant's SSC.
- For each relevant SSC, determine from the existing safety / risk studies its »seismic margin«, as discussed under the approach to core damage margin evaluation.
- Based on the individual seismic margins for relevant SSCs, determine the representative seismic margin for each critical safety function, considering the same general rules which were applied for core damage evaluation.
- Evaluate the availability of all containment integrity success paths following a postulated seismic event, with increasing severity. Start with the lowest seismic events and gradually increase the severity, in terms of PGA. The point (seismic level) at which the last success path is disabled can be considered a »seismic margin« for the considered release category.

Identification of success paths for containment integrity

For early releases, one success path is defined: success of containment isolation. In terms of seismic response this refers to the function and integrity of isolation valves and containment penetrations.

Additional requirement for success, which is assumed implicitly, is success of containment structure to remain intact following an earthquake.

This success path (i.e. containment isolation) applies to all initiator categories considered for seismically induced core damage (i.e. LOCAs, Loss of ESW, and others) with addition of beyond design basis reactor vessel failure.

Also, additional success path, which is assumed implicitly, is prevention of core damage.

Accordingly, an early radioactivity release (following a seismic event) is prevented if core damage is avoided or if containment structure integrity remains intact and containment isolation is performed successfully.

For late releases, two success paths are defined. The first one is successful operation of Reactor Containment Fan Coolers (RCFC). The second one is successful operation of Containment Spray (CI) Recirculation in combination with LPSI Recirculation through heat exchangers.

These two success paths are only relevant if early release was avoided (i.e. no seismically induced containment structural failures and successful containment isolation).

The two success paths apply to all initiator categories considered for seismically induced core damage, plus beyond design basis reactor vessel failure, with the exception of induced total loss of ESW. In this last case, the only success path is avoidance of core damage. However, the strategies exist to limit any release.

As earlier, an additional success path, which is assumed implicitly, is prevention of core damage.

As in the evaluation of core damage margin, each critical function from every success path needs to be mapped to plant's SSCs, considering also the dependencies among the frontline and support systems.

Evaluation of the seismic margin for containment integrity

As with core damage margin, the margin for the containment integrity function is assessed by evaluating the availability of all success paths following a postulated seismic event, with increasing severity.

The evaluation of containment integrity seismic margin starts with the lowest seismic events and continues with gradual increasing of the seismic severity, in terms of PGA.

Based on the plant specific seismic fragility analyses, the expected containment response would be as follows.

Earthquakes in the range of PGA < 0.45 g.

The expected sequence in this seismic range can be bounded by a LOOP without additional failures of safety related SSCs. (For more details, refer to the evaluation of seismic margin for core damage.) Success path for early releases is not challenged as seismic capacities for containment structure and containment isolation are both above 1 g, with high confidence. Also, neither of the two success paths for late releases is challenged in this range.

Earthquakes in the range of 0.45 g < PGA < 0.60 g.

Containment structure or isolation function is not challenged in this range (i.e. early release is considered a low probability event). The expected sequence in the range is a LOOP with possible, although not likely, additional failure of CST and / or RWST (failure probability at the upper end < 5%). The transfer of the RWST inventory into the containment is required for the second success path for late releases (i.e. for containment heat removal by combined CI / LP ECCS recirculation). (Note: The expected sequence in this range is not any of the LOCA or SLB sequences which would directly require the RWST for core damage prevention (due to the high seismic capacities of underlying SSCs). The expected sequence is LOOP which could either be converted into a LOCA sequence (attempted primary feed and bleed) or develop into high pressure core damage and reactor vessel failure. In the latter case, the transfer of RWST inventory into the containment would be performed post core damage.) Even if the RWST structural failure occurs, the transfer of cold water into the containment sump can be performed by some of alternative ways described in SAG-6 (see details in chapter 6.2.1.2.2). The remaining functions in the second success path for late releases (i.e. CI and LP ECCS) share a number of support systems with the first success path for late releases (i.e. RCFC). The controlling seismic failure mode, based on the plant specific seismic fragility analyses, is failure of EDGs. However, in this seismic range, the seismic failure of EDG is considered a low probability event. To summarize the discussion: Both success paths regarding the late releases are considered to apply.

Earthquakes in the range of 0.60 g < PGA < 0.75 g.

Containment structure or isolation function is not challenged in this range (i.e. early release is considered a low probability event). Seismic failure of RWST, however, cannot be excluded. In the case of RWST failure, the transfer of cold water inventory to the containment sump can be performed by some of alternative means described in SAG-6 (see details in chapter 6.2.1.2.2). The EDG (electrical periphery) is a controlling seismic failure mode for both success paths for late releases. Based on its seismic capacity, the failure of EDG is not considered likely in this range (failure probability at the upper end < 6%). It needs to be pointed that even in the case that EDGs fail, success path regarding the core damage would still be achieved by alternative means for secondary heat sink and alternative power supply to CVCS PDP for ensuring the RCS inventory and RCP seals integrity. (For details, refer to the evaluation of margin for core damage.) Therefore, even with success paths functions for prevention of late releases unavailable (which is not considered likely), there would be no releases. Regarding the other critical functions, the ESW structure failure is considered a low probability event in this interval. Functions of CI and LP ECCS systems are not considered to be challenged (seismic capacity > 1 g, with high confidence).

Earthquakes in the range of 0.75 g < PGA < 1.0 g.

Containment structure or isolation function is still not challenged. At this interval, however, the seismic failure of EDGs is considered likely (at 0.85 g, probability ≤ 10%). Assuming the failure of EDGs, both CI and LP ECCS would be unavailable (and so would the success paths regarding late releases). Failure of ESW Pump House structure would not have an additional impact (assuming the EDGs failure). Thus, at lower end of the seismic range, the expected sequence is seismic SBO with unavailable CST and RWST. Toward the upper end of the range, there is increased possibility of seismic ATWS with SBO conditions. (Refer to the evaluation of core damage margin.) The point where seismic core damage would become likely is assessed to be at 0.8 g or higher. As long as core damage is avoided, there would be no (late) releases. However, it needs to be pointed that even for the seismic event with reactor core damage and functions necessary for late release success paths (i.e. RCFC and CI / LP ECCS) unavailable, there are still strategies for controlling containment conditions, defined in SAG-6 (see details in chapter 6.2.1.2.2). Those strategies would ensure slow containment heat up rate and would enable performing planned and scrubbed releases.

Earthquakes in the range of PGA > 1.0 g.

At seismic levels of, approximately, 1 g, a number of SSCs is expected to fail, including CST, RWST, EDGs and ESW. Certain degradation of fuel assemblies' geometry in the core is also expected, which can prevent the control rods to drop in the core, causing the reactor scram failure. Core damage is considered unavoidable. For more details, refer to the evaluation of core damage margin. Regarding the seismic capacities of containment structures, the controlling failure mode would be the shield building. Assuming possible collapse of shield building structure, integrity of containment penetrations cannot be credited. Having in mind the seismic capacity, the seismic event at which early release (due to structural failures) would be likely is considered to be values significantly higher than 1 g. For example, structural failure probability for the shield building can approach 10% at levels 1.6 g to 1.7 g. Regarding the late releases (which are considered to be relevant when there is no early releases), the following conclusion applies: As long as buildings structures and geometry is preserved, the late releases can be, to certain degree, controlled by the SAG-6 (see details in chapter 6.2.1.2.2) strategies, noted above (timing, scrubbing, planning of the releases).

Conclusion regarding the seismic margin for containment integrity – cliff edge effect

Early Releases

Seismic events at which early radioactivity releases into the environment would be likely to occur are considered to be of PGA significantly exceeding 1 g. At these seismic levels, the collapse of shield building cannot be excluded. Under such circumstances, the integrity of containment isolation paths cannot be credited.

Late Releases

Seismic events at which late radioactivity releases into the environment would be likely to occur are considered to be of PGA in the range of 0.8 g or higher. This estimate is dictated by the fact that core damage is considered likely at this range of seismic events. It would occur under conditions where, likely, neither EDGs nor ESW / CCW would be available. This would, in turn, mean that containment heat removal functions (RCFC or CI / LP ECCS) are not available. Therefore, certain late release needs to be assumed under such circumstances. However, it needs to be pointed out that there would still be strategies for controlling containment conditions, defined in SAMGs (see details in chapter 6.2.1.2.2). Those strategies would ensure that any release to the environment is limited. The releases are expected to be scrubbed and planned.

Spent fuel pool

For Spent fuel pool (SFP), the success path can be formulated as:

- Structural integrity is preserved, and
- SFP water inventory is maintained.

The SFP water inventory can be maintained by:

- Maintaining the water inventory sub-cooled with respect to the boiling point (by means of the SFP cooling system), or
- Providing makeup water for the water lost due to the boiling and preserving the water level in the pool.

Seismic risk analyses were typically focused on the core damage risk. The risk from the SFP was considered low on account of long boil-off time (of the order of 3 days or more, which, of course, assumes that integrity of the pool is preserved).

It was similar with Krško SPSA. The Fuel Handling Building was not evaluated as no equipment which is essential for safe shutdown of the reactor is located in this building.

Generic study described in NUREG/CR-5176 (Seismic failure and cask drop analysis of the spent fuel pools at two representative nuclear power plants) was done in support of the resolution of Generic Issue 82 »Beyond Design Basis Accidents in Spent Fuel Pools«. This study used two representative spent fuel pools – a PWR and a BWR. These pools have been designed to meet the seismic design criteria existing in the late 1960s. It was found that the SFP structure, which is designed to retain large amounts of water and to withstand gravity and lateral loads from the fuel racks has a relatively high seismic capacity: the HCLPF capacity of the pool structure was estimated to be more than 3 times the SSE value in both cases.

The assessment of potential failure modes of the fuel pool racks and fuel assemblies has indicated that the fuel rack design is such that the assembly cannot be compressed into a critical mass thereby leading to a severe accident.

Parts of the cooling and makeup system for SFP are not designed as seismic class and as such failure of some components might be possible at relatively low seismic levels. However, the failure of cooling and makeup systems would not uncover the spent fuel assemblies for more than 3 days; it is expected that some recovery actions could be taken in this time period.

The above results and conclusions are considered applicable to Krško NPP SFP, as Krško NPP was designed and constructed in accordance with design requirements for US plants. Based on such reasoning (and in the absence of plant specific studies), it is likely that the integrity of NEK SFP would be challenged at PGA range of 0.9 g or higher The evaluation of seismic margin for the SFP at NPP is then summarized as follows.

Earthquakes in the range below the OBE (PGA < 0.15 g).

Complete LOOP is considered of low probability even at the upper end. Power supply would be transferred to 110 kV. The SFP cooling system is expected to continue normal operation. Success path is not considered to be challenged.

Earthquakes in the range between the OBE and SSE (0.15 g < PGA < 0.30 g).

In the upper part of the range, normal operation of SFP cooling system cannot be credited. However, the alternative strategies described in the EOP Appendix 33 would be implemented to provide the makeup water for the maintenance of SFP water inventory. The SFP integrity is not challenged. Success path for the SFP cooling would be supported by alternative strategies described in the EOP Appendix 33.

Earthquakes in the range between the SSE (0.30 g) and around 0.9 g.

Normal SFP cooling system operation cannot be credited. According to the EOP ECA-0.0 Appendix 33, the time to uncover fuel assemblies is 76 hours. It is expected that during this time the alternative strategies for SFP water inventory makeup, described in the ECA-0.0 App. 33 and in SAMGs (see details in chapter 6.2.1.2.2) would be implemented, which would enable long term cooling of the SFP. Parts of the falling objects may mechanically damage the fuel assemblies, but they are not expected to degrade the fuel assemblies matrix to the point that it would be uncoolable. The SFP integrity is expected not to be challenged, based on the generic study. Alternative strategies from ECA-0.0 (App. 33) and SAMGs (see details in chapter 6.2.1.2.2) are credited to provide the makeup water for the SFP inventory.

Earthquakes in the range of PGA > 0.9 g).

In this range gross structural failures of SFP cannot be excluded. Fuel damage can be expected.

Conclusion regarding the SFP integrity – cliff edge effect

For earthquake levels up to, approximately, 0.9 g, it is considered that the SFP integrity would not be challenged. Alternative strategies from ECA-0.0 (App. 33) and SAMGs (see details in chapter 6.2.1.2.2) are credited to provide the makeup water for the SFP inventory and, thus, prevent the fuel assemblies from overheating in the case of the small leakages or loss of inventory during evaporation.

Accordingly, for earthquakes in the range of PGA exceeding 0.9 g, gross structural failures of SFP cannot be excluded. For earthquakes of such intensity it can be expected that fuel uncovery in the SFP would occur.

2.2.3 Vulnerability of onsite and offsite emergency equipment and emergency operator facilities

All emergency equipment is now sited in the yard of the Krško NPP. In the future it will be located in the special building which will be designed with two times the design basis earthquake value (2×SSE).

Regarding the emergency operator facilities, the plant uses the following emergency facilities:

- technical support center (TSC) – it is a part of seismic inland, i.e. control building; HCLPF: 1.18 g),
- operational support center (OSC) – it is not a part of seismic inland (located onsite),
- offsite part of Emergency response organization (ERO), i.e. Emergency operations facility (EOF), which is activated in case of site emergency or general emergency; EOF is located about 100 km from the plant in Ljubljana City, which is also the location of Civil Protection Headquarters of the Republic of Slovenia,
- other emergency facilities (e.g. fire building (HCLPF:0.15 g), security building, etc.) are also not part of seismic inland.

2.2.4 Range of earthquake exceeding DBE and potential consequent flooding exceeding DBF

2.2.4.1 Physical impact

For the consideration of seismically induced floods, relevant are hydro power plant (HPP) dams at the Sava river. Additionally, potential formation of a natural dam (and its subsequent failure) following a catastrophic earthquake needs to be considered.

There are a number of studies and analyses related to the flooding hazard induced by failures of hydro power plant dams at Sava river. HPPs at Sava river, which are relevant for NPP due to potential flooding safety implications, are divided into two groups: there is a group of operating HPPs at the upper Sava river and, at the lower Sava river, there is group of hydro plants which are in different stages of operation, construction and planning. The region of upper Sava and the region of lower Sava are different seismic regions and it is not considered that a single seismic event could have a damaging impact on both groups.

The group of HPPs at upper Sava river is comprised of HPP Moste, HPP Mavčiče and HPP Medvode. They are all operating plants.

The group of HPPs at lower Sava river is comprised of:

- HPP Vrhovo, HPP Boštanj, HPP Blanca, which are operating plants;
- HPP Krško, which is under construction;
- HPP Brežice and HPP Mokrice, which are in different stages of design or planning.

The HPPs Vrhovo, Boštanj, Blanca and Krško are (will be) located upstream of NPP. The HPPs Brežice and Mokrice will be located downstream of NPP.

A number of studies have been performed related to the failures of dams upstream of NPP, with different scenarios considered. When developing the scenarios, the requirements from the standard ANSI/ANS-2.8-1992 were considered.

Plant specific analyses included postulated damage of all three HPP dams at upper Sava river, with assumed initial presence of 25-yr flow along the whole river. The resulting flood wave would, at the region of lower Sava river, have a peak in the range from 100-yr flow and 1000-yr flow. It would cause, downstream of HPP Boštanj, considerable flooding, mostly in the area of Dolenji Boštanj and Sevnica, but also in the area downstream of NPP. However, from the description of flooded condition, it can be seen the considered flood would not have a safety impact on the NPP. For the details on impact of 100-yr and 1000-yr flood, refer to section 3.2.

For the group of HPP dams at lower Sava, a number of scenarios were developed and analyzed. The basic critical scenarios considered for particular area were:

- Cascading failure of one dam gate at the upstream HPPs, with all the gates at the first downstream HPP blocked (closed);
- Failure of all gates at the first upstream HPP, with all the gates at the first downstream HPP blocked (closed).

None of the scenarios was found which would threaten the safety of the NPP. (Actually, at the current status, only the HPPs Vrhovo and Boštanj are relevant for the NPP.) With assumed 25-yr flow as initial conditions, all the scenarios were found to be less severe than the scenario with failure of all three dams at upper Sava.

Additionally, the scenario with failure of all gates at the HPP Vrhovo with simultaneous opening of all gates at the HPP Boštanj was analyzed. The results of calculation showed the flood wave with peak at 2257 m^3/s and duration of 8 h. As the peak of the flood wave is below the 100-yr flow, it can be concluded that assumed scenario would not threaten NPP or its surroundings. For the details on impact of 100-yr flood, refer to section 3.2. The consequence of such an event would be flooding of the area downstream of NPP protective dikes.

Additionally to the HPP dam failures, recent study investigated the risk from the formation of a natural dam (and its subsequent failure) after a catastrophic landslide or large rock fall, following an earthquake. A case of forming of a natural dam is possible only during a catastrophic earthquake that could lead to a slope failure of extreme dimensions.

It is necessary to consider what can be said about the maximum (peak) discharge that happens immediately downstream of the damming. The maximum discharge decreases with the distance from the damming into the downstream direction due to the flattening of the dam-break flood wave. A scenario for the forming of a large lake (behind the natural dam) is only possible during a very strong earthquake that would trigger a large debris flow, a landslide or a rock fall. This would, according to the plant specific study, require an earthquake that is of the 9^{th} or 10^{th} grade on the European Macroseismic Scale (EMS) scale, or an earthquake with the magnitude well above 6. In terms of PGA, this would mean an earthquake in the range of 0.6 g or higher. The study evaluated potential consequences of a large debris flow, a landslide or a rock fall resulting from such an earthquake. Regarding the first, it was estimated that the critical events that would be a consequence of a debris flow pose no threat to the Krško NPP, especially due to its location in a safe distance from a potential debris-flow source area. As for the induced landslides, the examination of the regional geological setting showed that critical events as a consequence of a landslide would not threaten the Krško NPP. Looking at a critical event which could be triggered by a rock fall, it was determined that in the area before the Sava river enters the Krško-Brežice Basin, small rock falls from steep slopes are possible, but they cannot dam the Sava river.

2.2.4.2 Weak points and cliff edge effects

The details on the plant response and weak points regarding the flooding from Sava river are provided in Section 3.2.2. However it has to be pointed out that no cliff edge effects have been identified.

2.2.4.3 Measures which can be envisaged to increase robustness of the installation

The envisaged measures to increase robustness of the NPP against the external flooding hazard are discussed in section 3.2.3.

Regarding additional protection against the external flooding from Sava river the improvement under implementation - increasing the elevations of dikes, will be beneficial.

3 Flooding

3.1 Design basis

3.1.1 Flooding against which the plants are designed

3.1.1.1 Characteristics of the design basis flood (DBF)

Design flood

Protection from floods of 0.01% frequency, which represents the statistically calculated 10000-year flood, was accomplished by the plant design. Design flow of 10000-year flood is calculated as 4790 m^3/s and corresponds to the 155.35 meters above Adriatic Sea level (m.a.A.s.l.) water level at the dike. The plant site elevation is 155.20 m.a.A.s.l. Plant buildings have entrances and openings at the altitude of 155.50 m.a.A.s.l. providing the corresponding safety degree for plant structures even without or totally failed Sava river protection dike.

Probable maximum flood

Beside the design flood (10000-year flood), the plant is protected against the occurrence of the probable maximum flood (PMF) which represents the hypothetical flood that is considered to be the most severe reasonably possible, based on application of probable maximum precipitation and other hydrologic factors favorable for maximum flood runoff such as sequential storms and snowmelt. Protection against PMF is achieved with appropriate design of interface structures and with the protection dike. Probable maximum flood flow is 6500 m^3/s and corresponds to the 155.89 m.a.A.s.l. water level at the dike. Regarding the Sava river flooding behavior, the nuclear power plant will not be endangered by the occurrence of the probable maximum flow, provided that evacuation of greater quantities of water is ensured via the right inundation (the NPP is located on the left Sava river bank). The design flood elevation of the interfacing structures is 156.50 m.a.A.s.l., considering 0.6 m of safety margin the elevation of the structures plateau is located at 157.10 m.a.A.s.l. The margin elevation also serves as the protection against wind activity. Under supposition that wind velocity is 21 m/s from most adverse north-west direction at which it is possible to assume the effected length of activity 900 m, a wave of 0.93 m might be formed. Consequently, it is possible to expect raising of water surface at the nuclear power plant for 0.46 m, resp. at the elevation 156.35 m.a.A.s.l, still providing 0.15 m safety margin considering design flood and 0.75 m safety margin regarding the structures plateau elevation.

Local heavy rainstorm

Flash floods due to local heavy rainstorms can in general pose a hazard to nuclear power plants. The comparison of the runoff of the water from the site with the maximum hourly intensity of rainfall in the local area gives the potential water level increase around the buildings in case of higher rainfall that runoff.

The detailed analyses show that the height of the standing water after the first hour would be 29.25 mm, with the conservative assumption that this water will not evapotranspire or seep into the soil. As average rainfall intensity over periods longer than one hour is lower, the height of standing water will decrease after the first hour. Structures entrances are at least 300 mm above the soil level, so the level of standing water does not represent a threat to the systems.

Krško site is located on higher ground elevation than surrounding area, which provides the capabilities for natural water drainage, by gravity.

3.1.1.2 Methodology used to evaluate the design basis flood

Design flood

The time series of maximum annual Sava flows in Krško, originating from the period 1926 – 2000 represent a statistical flood population sample that should be described by a theoretical distribution functions.

Table 1: Maximum Sava river flow in Krško [m³/s]; Period: 1926 – 2000

Year	Flow m^3/s	Year	Flow m^3/s	Year	Flow m^3/s	Year	Flow m^3/s
1926	2118	1946	713	1966	2357	1986	1501
1927	2165	1947	1740	1967	1694	1987	1897
1928	1346	1948	2005	1968	1773	1988	1358
1929	893	1949	1881	1969	1941	1989	1412
1930	2165	1950	1016	1970	1683	1990	3050
1931	1735	1951	1685	1971	1073	1991	1947
1932	1413	1952	1863	1972	2116	1992	2169
1933	2940	1953	1401	1973	2549	1993	1840
1934	1998	1954	1914	1974	2382	1994	1591
1935	1586	1955	1098	1975	1992	1995	1425
1936	2231	1956	1820	1976	1452	1996	1867
1937	1652	1957	1134	1977	1448	1997	1252
1938	1531	1958	1770	1978	1225	1998	3001
1939	1789	1959	2009	1979	2630	1999	1125
1940	2275	1960	1531	1980	2474	2000	2080
1941	934	1961	2044	1981	949		
1942	813	1962	1836	1982	2338		
1943	1798	1963	1736	1983	1467		
1944	1366	1964	2733	1984	1456		
1945	847	1965	2001	1985	1910		

Selection of the theoretical distribution function of the Sava maximum flows in Krško shall be defined by six functions that are usually used for evaluation of flood occurrence probability.

These functions are:

- Normal function
- Log Normal function
- Pearson function, type III
- Log Pearson function, type III
- Gamma function (two parameter)
- Gumbel extreme values function.

By considering the statistical criteria and the trend of data of floods with return periods of 5 years and more, Gamma distribution is chosen as the most qualitative one and thus the 10000-year flow of the Sava river at Krško for the period 1926 - 2000 is 4431 m³/s. To be on the safe side we may choose Log Normal distribution with the 10000-year flow of the Sava river at Krško of about 4700 m³/s.

The national value for 10000-year flood was calculated by the Institute for Water of the Republic of Slovenia using Log Pearson III distribution and the result is the flow of 4790 m^3/s. Krško NPP conservatively chose this flow as the value of 10000-year design flood.

Probable maximum flood

Determination of the probable maximum flow (PMF) is characterized by the following steps:

a. determination of probable maximum precipitation (PMP)
b. determination of runoff loses with runoff model
c. determination of probable maximum flow and level
d. casual waves' activity due to wind
e. potential danger due to upstream dams failure (seismic origin)

The probable maximum precipitation is »the theoretically highest amount of precipitation that has fallen down during a certain period, which is physically possible, over a certain drainage area«. The mean probable maximum precipitation on the Sava drainage area of 7850 km^2 was determined as 365.1 mm, with the flow of 6500 m^3/s and the volume of the flood wave amounting to 2866 million m^3. From the flood wave shape the maximum rate for flood increase was determined as 140 $m^3/s/h$.

Probable maximum flow 6500 m^3/s corresponds to the 155.89 m.a.A.s.l. water level at the Krško NPP dike. The nuclear power plant will not be endangered by the occurrence of the probable maximum flow, provided that evacuation of greater quantities of water is ensured via the right river bank inundation.

Due to wind activity with the wind velocity of 21 m/s from most adverse north-west direction at which it is possible to assume the effected length of activity of 900 m, a wave might be formed with raising of water surface at the nuclear power plant for 0.46 m reaching the elevation 156.35 m.a.A.s.l. However, there is still 0.15 m safety margin to the design flood and 0.75 m safety margin to the interface structures plateau elevation.

At the flood situation of probable maximum flood, the upstream dams would have no effect as there are only small-dam hydropower plants and would have all gates already opened by the occurrence of PMF.

Flood wave caused by dam failures

For the upstream hydropower plants the analysis supposes that the breaking of dams is instantaneous and complete caused by earthquake, and that this happens at the moment when the upstream storages are full, while the Sava flow corresponds to the occurrence of 25-year flood.

The mathematical model obtained by the method of ultimate recharges shows that the flood wave amounts to 3333 m^3/s. The safety of nuclear power plant would not be endangered by the breaching of the existing hydropower plant structures.

Local heavy rainstorm

The extreme value distribution was fitted to maximum hourly rainfalls near Krško NPP from 1970 to 1986. Runoff Analysis of water accumulated due to rain was calculated according to a method to determine the volume of water from a rainstorm that must be removed (i.e., water that does not evapotranspire or seep into the ground).

The runoff coefficient takes into account the type of soil. The height of the standing water left after the first hour that did not evapotranspire or seep into the soil was determined as 29.25 mm. As average rainfall intensity over periods longer than one hour is lower, the height of standing water will decrease after the first hour. Structures entrances are 300 mm above the soil level, so the level of standing water does not represent a threat to the systems. Krško NPP site is located on higher ground elevation than surrounding area, which provides the capabilities for natural water drainage, by gravity.

3.1.1.3 Conclusion on the adequacy of protection against external flooding

Plant building entrances and openings are constructed above the elevation of the 10000-year flood. So the plant is safe for the occurrence of the design flood.

Plant is also protected against the probable maximum flood (PMF) with the appropriate design of the Sava river interface structures and with the protection dike for protection of plant site against probable maximum flood with additional waving due to winds. With PMF flood level at 155.89 m.a.A.s.l and additional waving it is possible to expect rising of water level at the nuclear power plant for 0.46 m to the elevation 156.35 m.a.A.s.l, still providing 0.15 m safety margin considering design flood and 0.75 m safety margin regarding the interface structures plateau elevation. However, as the nature phenomena changes with the time, the adequacy must be reevaluated every decade or so, which is the regular practice in Krško NPP.

Flood wave caused by hydropower plants dam failures is not a threat to the Krško NPP since maximum flood wave would reach only the flow of 3333 m^3/s what is much less than the design flood wave.

Heavy local rainstorm does not jeopardize the safety of the plant as NPP Krško site is located on higher ground elevation than surrounding area, which provides the capabilities for natural water drainage, by gravity.

3.1.2 Provisions to protect the plant against the design basis flood

3.1.2.1 Key structures, systems and components (SSC) required for achieving safe shutdown state and supposed to remain available after the flooding

3.1.2.1.1 Provisions to maintain the water intake function (where applicable)

The Essential Service Water system (ESW) provides cooling water to the component cooling system and boron thermal regeneration system to transfer the plant heat loads from these systems to the ultimate heat sink, the Sava river. The system also serves as a backup safety related source of water for the auxiliary feedwater system (AFW). The component cooling system (CCW) provides cooling for safety systems and engineered safety feature systems. Therefore, the essential service water system is a safety related system with isolation capabilities and redundancy in components and features necessary to satisfy the requirements of these systems under normal or accident conditions combined with a loss of offsite power and any single active or passive failure.

Water cleaning equipment prevents the plugging of water intake function. One safety class, automatic, self-cleaning strainer is installed in the discharge line of each pump. Each strainer is equipped with a safety class backwash line for debris removal. The strainer and backwash line are in operation whenever the associated essential service water pump is operating.

The intake structure incorporates redundant inlets and channels for supplying river water to the pumps. Each channel includes a movable trash rake followed by a traveling water screen. The two channels are connected to a common forebay serving the three pumps. The inlets, channels and traveling screens are adequately sized for handling the flow for both loops (plus the fire service water pumps, which are also located in the intake structure). Thus, the intake structure provides complete redundancy in flow paths to the essential service water pumps. Water for the traveling screen wash systems is supplied from the essential service water loops.

In case of flooding, operating provision of frequently monitoring of intake systems is provided, to assure the prompt actions in case of high river pollution which may occur during the flooding event, to maintain the water intake function.

The ESW is designed for operation with any water level varying from the original minimum river level, at an elevation of 147.85 m, to a maximum flood level at an elevation of 156.60 m. However if in some rare case the ESW system would fail to provide the water intake function due to common cause, as emergency provision, the AFW system may be used to provide cooling with on site tank capacities of cooling water to enable the cooling function until the restoration of the ESW system.

3.1.2.1.2 Provisions to maintain emergency electrical power supply

There are two safety emergency diesel generators on the plant and two independent offsite power systems available to deliver power to the plant.

The switchyard and power lines are located on the left bank of Sava river protected from flood.

However, if the probable maximum flood would occur this could result in nationwide power availability problem due to extensive raining, wide area flooding and landslides. So, it is a reasonable possibility that offsite power might become unavailable in case of probable maximum flood.

When offsite power becomes unavailable, Krško NPP has the 110 kV dedicated power line over the hill to the nearby Brestanica thermo power plant which has a procedure to deliver the power solely to Krško NPP in cases of nationwide power loss.

In case of loss of offsite power, two independent safety emergency diesel generators are available on the site, located within diesel generator building at elevation of 155.50 m.a.A.s.l. They are not influenced by design flood and probable maximum flood. The cooling of the diesel generator engines is independent from Sava river.

3.1.2.2 Main associated design/construction provisions

Main provision against design basis flood is the design elevation of plant structures and openings. Plant site plateau is located at elevation of 155.20 m.a.A.s.l., while the openings in structures are located at elevation of 155.50 m.a.A.s.l. The design flood elevation is 155.35 m.a.A.s.l. The portions of Safety Class structures located below 155.20 m.a.A.s.l. are protected on their outside surfaces by a continuous waterproofing membrane. Water stops are provided at construction joints. Any potential in-leakage from such phenomena as cracks in the structure walls and leaking water stops is collected in sumps and pumped out.

Provisions against the probable maximum flood are fulfilled by appropriate design of interface structures and by protection dike. With PMF flood level at 155.89 m.a.A.s.l. and additional waving of 0.46 m the water level would reach 156.35 m.a.A.s.l.

At cooling tower exhaust to circulating water system, stop logs are inserted in case of Sava river reaching the elevation of 155.70 m.a.A.s.l., to prevent water irruption from cooling tower water canal at elevation of 155.90 m.a.A.s.l.

3.1.2.3 Main operating provisions

Several procedural steps are referring to operating provisions in case of increased Sava river flow. At flow 700 m³/s increased monitoring of Sava river flow, elevation, pollution and cooling systems begins. Frequent monitoring of intake systems is provided, to assure the prompt actions in case of high river pollution which may occur during the flooding event, to maintain the water intake function

Before Sava river reaching the elevation of 152.50 m.a.A.s.l. drainage system is set to stand-by status and regular exhaust to Sava is closed to prevent backflow and water irruption.

When Sava river rises to the elevation of 155.70 m.a.A.s.l, stop logs are inserted at cooling tower exhaust to circulating water system.

Regular survey of protection dikes on the 5 year basis is provided by a surveillance procedure. Obstacles removing and mowing is performed on one year basis. Unplanned review of dike status is initiated in cases of seismic activity, in case of flooding and in case of land works near or on the dikes. The survey of the dikes includes obstacles removing and mowing, review of stability condition of the dikes through the tests of strength and deformation properties, tests of soil stability, search for cracks, soil movements and water penetration, review of drainage availability and stability, review of dike erosion.

Measurement of the Sava river cross sections from 4 km upstream to 15 km downstream is provided on the 5 year basis or after a major flood. The activities include also correction of maps and eventual corrective actions in case of dangerous erosion.

3.1.2.4 Other effects of the flooding taken into account

3.1.2.4.1 Loss of external power supply

See section 3.1.2.1.2.

3.1.2.4.2 Loss of water intake (effects of debris, oil slicks, etc.)

On the Sava river, Krško NPP has two input structures, one for condenser cooling for power operation and a separate one for cooling of safety relevant equipment (the ESW; see 3.1.2.1.1).

The Circulating Water (CW) system provides cooling water for the main condenser and turbine building auxiliary coolers, 25 m³/s of river water is used to remove waste heat from the energy cycle, which is adequate to satisfy system requirements for all normal and abnormal conditions including turbine trip from full load. The heated circulating water can be discharged to the river or partially recycled through a cooling tower. The portion of heated water not recycled is discharged to the river. The intake structure located on the river bank contains, in addition to the circulating water pumps, six traveling water screens, a traveling water trash rack, screen wash pumps, strainers, and auxiliary service equipment.

Additional procedural steps are taken in case of expected or unexpected pollution of the Sava river. If procedural measures are ineffective, reactor power shall be lowered or even reactor shutdown must be initiated.

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

By the occurrence of the design flow and by the occurrence of the probable maximum flow, the flooding would partially flood the road from the plant to the safe area. The flooding event, according to the recorded history the longest flooding event would last around 17 hours. The plant is not directly jeopardized by the probable maximum flood and the personnel present at the plant could operate the plant.

At the flow of 3765 m³/s, the plant alert is initiated and technical support center and operations support center are called to the plant site. At flow of 4274 m³/s, site emergency would be initiated and offsite emergency operations facility in Ljubljana would be activated. At this flow all roads to the Krško NPP are still available.

At flows higher than PMF the access to the plant site would be unavailable and offsite alternative flooding-safe location in Videm for technical support center and operations support center would be activated. The intervention stuff has telephone and radio communication available at this alternative location. In case of need, they would organize civil protection forces and professional fire brigade to gain the access to the plant.

3.1.3 Plants compliance with its current licensing basis

3.1.3.1 Operator's general organization to ensure conformity

Refer to section 2.1.3.1.

3.1.3.2 Operator's organization for mobile equipment and supplies

Refer to section 2.1.3.2.

3.1.3.3 Deviations and remediation actions

During the history of NPP operation there were three significant floods in the area. Those were in the year 1990, 1998 and 2007. These events, together with the fact that there were several new hydro power plants (HPP) at the Sava river in the various stages of construction, design and planning, led to performance of a number of studies and analyses related to external flooding hazard for NPP, during the past decade.

A comprehensive study regarding the possible solutions for improving protection against external floods at NPP was done in 2005. This study, among other aspects, provided the update to the estimate of 10000-year flood, based on the updated history of maximum river flows, and confirmed the adequacy of NPP design requirements regarding the 10000-year return period.

The original design flow was 4272 m^3/s and the level of the Sava floods (10000-year flood) was originally estimated at the 156.27 m.a.A.s.l. Revised design flow is 4790 m^3/s and corresponds to the 155.35 m.a.A.s.l. (according to revised flow level relationship tables) water level at the dike.

The dikes around the plain exceed this level by 118 – 138 cm and outside of this plain, the crest elevation of the dike is 98 cm above PMF level.

Comparing the original data and revised results, the difference in the level results may be observed (flood flow increase from 4272 m^3/s to 4790 m^3/s but level due to flow decreased from 156.27 m.a.A.s.l. to 155.35 m.a.A.s.l.). The cause for the level decrease is in the assumptions for the original water level versus Sava flow relation. The valid assumption at that time was that the dike will be built on the right side of the Sava river from the NEK location to Čatež for population flood protection. The planned dike was never built. Lately the analyses were performed to obtain the reliable solution of the problem of the flood protection on the larger area, in our case upstream (3 km) of the plant dam to the point in Krško, where the Sava river leaves its canyon passing below the bridge, to the point 4 km downstream of the plant dam. The model was built with the 20 × 20 m resolution for this area. Computer software tool PCFLOW2D was used for dynamic water flow calculations.

The 2005 study also included the re-evaluation of the PMF, which ended with conclusion that the existing PMF estimate in USAR is still applicable. Revised probable maximum flood flow remained 6500 m^3/s and revised PMF level corresponds to the 155.89 m.a.A.s.l. water level at the dam.

This study was followed by a number of other studies aimed at performing a comprehensive evaluation of PMF and corresponding hydraulic analyses for the area surrounding the NPP. One of the purposes of those studies was to evaluate possible impacts of planned new HPPs at Sava river on flooding safety at NPP and to propose possible measures / options for additional improvements.

Among those, one of the most recent was a study which considered different conservative flooding scenarios designed on the basis of the applicable ANS Standard concerning the design basis flooding, ANSI/ANS-2.8-1992. This study represented a preparation of a conceptual design package for flood protection of Krško NPP.

Accompanying this study was another one, representing hydraulic analysis of high Sava river levels at NPP site for the revised probable maximum flood.

Additionally to this one, another hydraulic analysis of high Sava river levels at NPP site was performed (also for the revised PMF), with the purpose to provide projected options for additional flooding protection, considering also planned new HPPs at Sava river.

3.1.3.4 Specific compliance check already initiated by the licensee following Fukushima NPP accident

Floods were recalculated according to the conservative scenarios from ANSI/ANS-2.8-1992 in latest studies that demonstrate that at revised PMF flood of 7081 m³/s, the NEK plain would still be an island. All SSCs at the ground level would be safe. The underground water should have no impact on the safety SSCs. At the PMF flow margin of flood elevation still exists as the PMF flood level of Sava river is 156.41 m.a.A.s.l. and the level of dike is 157.10 m.a.A.s.l. On the left Sava river bank the flood level at the nearest point to the Krško NPP would be 154.43 m.a.A.s.l. and NPP plateau is at elevation of 155.20 m.a.A.s.l. and the NPP openings are at the level of 155.50 m.a.A.s.l. The update of licensing documents is currently in process and consequently the improvement of flood protection, to keep the left Sava river bank dry even for flows beyond the PMF flood flow is currently in process. On Figure 5 the situation at PMF flow of 7081 m³/s is presented with current status of protection dikes.

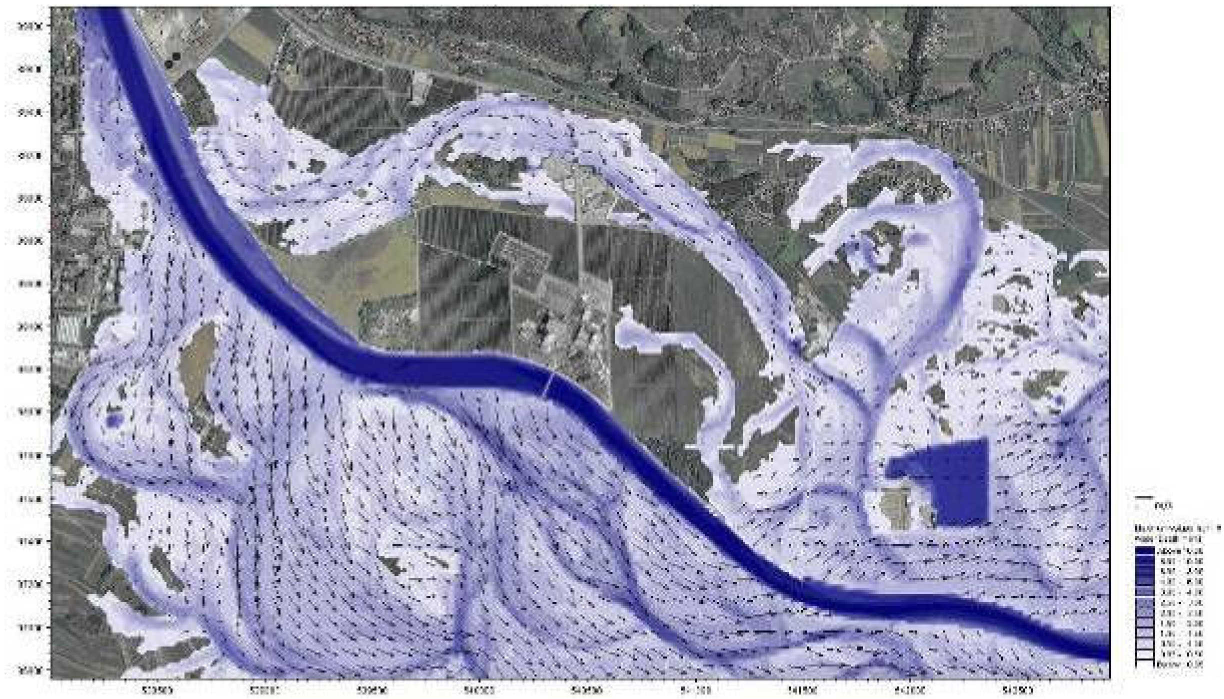


Figure 5: Situation at PMF flow of 7081 m³/s is presented with current status of protection dikes

After the competition of modification (increasing the dikes upstream of the plant), the flood at the PMF flow of 7081 m³/s is presented in Figure 6.

Figure 6: After the completion of modification (increasing the dikes upstream of the plant), the flood at the PMF flow of 7081 m³/s

Increased level of dikes upstream of the plant will be capable of protecting the plant to the flood flows beyond the PMF flood. The flood flow of 10000 m³/s is shown on the Figure 7.

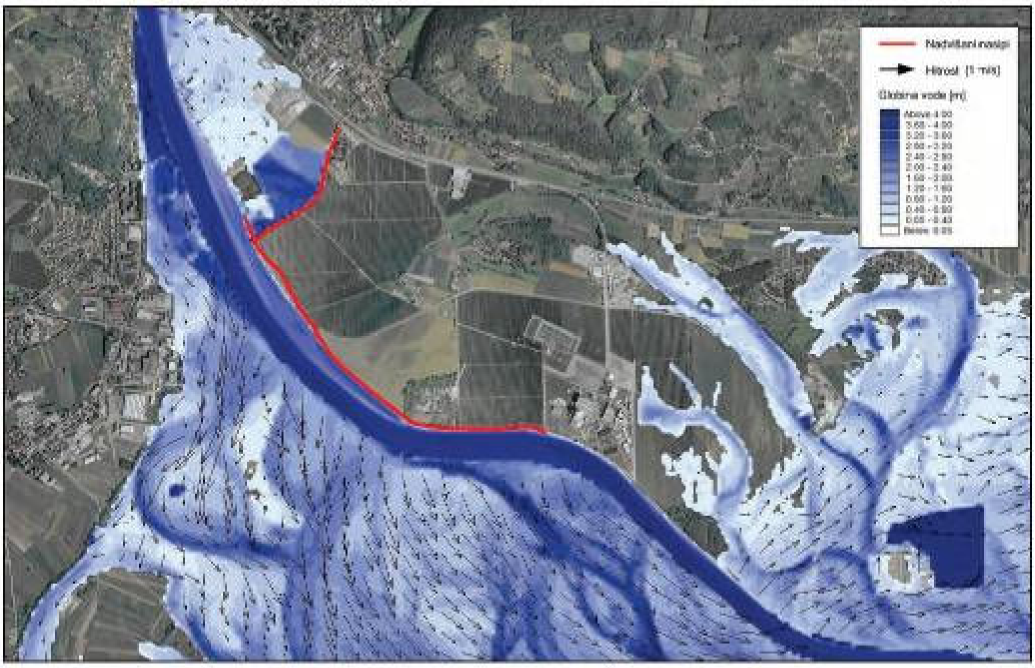


Figure 7: Increased level of dikes upstream of the plant will be capable of protecting the plant to the flood flows beyond the PMF flood. The flood flow of 10000 m³/s

3.2 Evaluation of safety margins against flooding

3.2.1 Additional protective measures which can be envisaged in the context of the design, based on the warning lead time

The conclusions from the plant specific studies concerning the flooding hazard are:

- The flooding event would be relatively slow in progression: The shortest time for developing the maximum flow would be 20 hours or more.
- The shape of flood wave determines the duration of the high flows which is not more than 24 hours. However, the discharge time of high water levels in the flooded area can be considerably longer.

The protective measures which can be envisaged based on the warning time would include implementing normal plant shutdown and verification of the status of the equipment necessary for the implementation of alternative methods for ensuring critical safety function, described in the EOP procedures.

3.2.2 Weak points and cliff edge effects

The evaluation of external flooding margins at NPP consists from the following main steps:

a. Use the USAR and available safety studies and analyses to:
b. Establish the relations between relevant elevations at NPP site and flooding water levels achieved at different river flows.
c. Identify the »success paths« for a range of flooding events. A »success path« is defined as a minimum set of functions required for avoiding reactor core damage state following a flooding event. Each success path identified is specified in terms of required critical safety functions.
d. Map each critical safety function in every success path to the specific plant's SSC.
e. Determine the location of each relevant SSC (e.g. building, elevation).
f. Evaluate the availability of all success paths following a postulated flooding event, with increasing severity. Start with the lowest flooding event and gradually increase the severity, in terms of a maximum river flow. The point (maximum river flow) at which the last success path is disabled can be considered an »external flooding margin« for the whole plant.

<u>Relevant elevations at NPP site and flooding water levels at different river flows</u>

The following relations between some characteristic elevations at NPP site and flooding water levels at different river flows can be established, based on the plant licensing documents and plant specific studies.

Water level (in the river) resulting from the revised (i.e. currently applicable) 10000-year flow is 155.35 m.a.A.S.l. This is currently applicable design flow level.

The Auxiliary Building (AB) and Diesel Generator Building (DGB) altitude or so called »NPP plain« is at the openings on 155.50 m.a.A.S.I.

Water level (in the river) which would be achieved by the PMF, as currently estimated in the plant licensing basis, is 155.89 m.a.A.S.I.

Water level (in the river) which would be achieved by the worst case scenario based on ANSI/ANS-2.8-1992 (according to the recent plant specific study) is 156 41.

Dike around the NPP plain is 157.10 m.a.A.S.I. The ESW and CW buildings are at the same elevation. Considering the NPP plain elevation of 155.50 m.a.A.S.I, the difference between the dike and the openings of buildings located on the plain is 157.10 – 155.50 = 1.60 m.

Based on the hydraulic analyses from the plant licensing documents and plant specific studies, the flooding impacts at characteristic river flows can be summarized as follows:

- At the 100-year river flow, all SSCs at NPP plain are completely safe. River flow would start spreading over the fields on the left bank, downstream of the dike and would start going backwards, in the »upstream« direction, behind the NPP.
- At flows of about 1000-year river flow value, the water would start to overflow the Potočnica creek dike. All NPP SSCs would be completely safe.
- At the 10000-year river flow, all NPP SSCs are completely safe. The water overflowing the Potočnica creek dike would flow downstream behind the NPP (following the route with lowest elevation) and join the water going backwards from south-east of NPP (from downstream of left bank dike). The overall water flow behind NPP plain would converge to downstream direction. The NPP plain would become an »island« between the river flow and inundated flow on the left bank.
- At flow corresponding to the current estimate of PMF in the plant licensing basis (6500 m^3/s), the most of the Krško Field would be under the water. The NPP plain would still be »dry« (an island) and all SSCs considered safe.
- At flows corresponding to the worst case of the conservative scenarios from ANSI/ANS-2.8-1992 considered in the plant specific study (7081 m^3/s), it was demonstrated that NPP plain would still be an island. All SSCs at the ground level would be safe. The underground water level should have no impact on the SSCs in the AB.

From the above considerations it can be concluded that if NPP plain is to be flooded, it would not happen from the river side, i.e. by water overflowing the dike. The incoming water would split and the NPP plain would form an island, which would be protected from the river side by the dike. The difference in elevations between the dike and the NPP plain buildings openings is 1.60 m. Flooding of NPP plain would come from behind.

Characterization of success paths

Success paths are defined separately for the case that flooding wave reaches the plant while at power and for the case that plant was normally shutdown. In both cases onset of LOOP conditions is assumed. Success paths can be summarized as follows.

In the case of LOOP with reactor at power, two success paths apply. Both of them require successful reactor trip and availability of onsite power.

The first success path is through the secondary heat sink and the RCS inventory. The required critical functions for the success are secondary heat sink (by means of AFW) and RCP seal injection (by means of CVCS Pumps). For the latter, there is an alternative success option: CCW to RCP Thermal Barriers.

The second success path is through the primary »feed and bleed«. The required critical functions for the success are operation of pressurizer PORVs and high pressure safety injection (HPSI) and recirculation. (It should be noted that the latter also involves low pressure SI function - LPSI pumps to provide suction for HPSI pumps.).

In the case of LOOP with reactor shutdown prior to the flood wave, the success path would be through the residual heat removal (RHR) function. The critical function required would be the RHR system in loop-to-loop operation. However, it needs to be pointed out that, in the case of RHR unavailability, alternative methods for decay heat removal would be implemented, in accordance with plant procedure ADP-1.3.030 Shutdown Safety. Those methods, as relevant for considered condition, would be secondary heat sink or primary feed and bleed. Therefore, similar considerations apply as for success paths for LOOP at power. The basic difference is that, if induced LOOP occurs with reactor shutdown, development of the sequence would be much slower. This would give additional chance to the alternative measures described in EOP ECA-0.0

Appendices to be effectively implemented even in the case that NPP plain is partially flooded or may give a chance to the flood wave to recede.

According to the described approach, the plant level external flooding margin is assessed by evaluating the availability of all success paths following a postulated flooding event, with increasing severity.

The evaluation of plant level external flooding margin starts with the lowest flooding events and continues with gradual increasing of the flooding severity, in terms of a maximum river flow.

Based on the plant specific analyses, the expected plant response would be as follows.

1. <u>River flow approaching 3290 m³/s (100-year flood).</u>

All SSCs at NPP plain are completely safe. Toward the upper end, the river flow would start spreading over the fields on the left bank, downstream of the dike. Normal operation continues with raised awareness of potential flooding hazard and need for monitoring and predicting the development of the hazard. All success paths for initiators from normal plant operation are available.

2. <u>River flow between 3290 m³ and 4040 m³/s (1000-year flood).</u>

All NPP SSCs are completely safe. Toward the upper end, the water would start to overflow the Potočnica creek dike. It is expected that within this flow range the decision would be made to shut down the plant, provided that there is no clear indication that the flow has reached its maximum. In this range, applicable are success paths for initiators from normal plant operation.

3. <u>River flow between 4040 m³/s and 4790 m³/s (10000-year flood).</u>

Toward the upper end, the water overflowing the Potočnica creek dike would join the water from downstream of NPP dike, cutting off the NPP plain, which would remain as an »island«. The plant is expected to be shut down. All NPP SSCs are completely safe. Success paths for initiators from normal plant operation at shut down modes are considered to apply.

4. <u>River flow between 4790 m³/s and 6500 m³/s (PMF in plant licensing basis).</u>

Toward the upper end, the most of the Krško Field would be under the water. The NPP plain would still be »dry« (an island). The plant would be shut down. All SSCs are considered to be considered safe. The upper end represents already extreme flooding conditions, not only for NPP, but for the whole area. At such conditions, LOOP can not be excluded. It would not be caused by conditions at NPP switchyard (except if the flood event is combined with extreme local weather). However, it may be caused by the grid conditions at other locations. Additional concern is that river flow may bring the debris which could clog the ESW intake. In such a case, the RHR system would not be available for decay heat removal (DHR). The DHR would be performed by alternative methods, according to the ADP-1.3.030, which would, in this case, mean the operation of AFW turbine driven pump (TDP). Following the depletion of CST, alternative means described in ECA 0.0 Appendices would be used. Success paths for initiators from normal shutdown conditions or, in the case of LOOP, success paths for LOOP apply. (In the case of LOOP, success path with RHR in loop-to-loop operation is expected to apply, since the plant is expected to be shutdown. In the case of loss of ESW due to clogging of intake structure, the DHR would be performed by alternative means, as described.)

5. River flow between 6500 m³/s and 7081 m³/s.

The upper end is considered to represent extreme flooding conditions at national level. The NPP plain would still be an island. All SSCs at the ground level are considered to be safe. The underground water is considered to have no impact on the SSCs in the AB. Plant is shutdown. Switchyard is dry. However, a LOOP cannot be excluded due to possible grid conditions at other locations. Regarding the potential loss of ESW due to clogging of intake structure, the same discussion applies as above. Either normal shutdown or LOOP condition success paths are applicable, as above.

6. River flow > 7081 m³/s.

It is not possible due to lack of specific hydraulic calculations to specify exactly the river flow at which the NPP plain would be completely flooded. Plant response strongly depends on the maximum water level and duration. Reactor core damage can be avoided at water flows significantly higher than 7081 m³/s. The first direct consequence of flooding the NPP plain is expected to be LOOP (if did not occur already), due to flooding of switchyard. However, the doors to safety related plant buildings are elevated by 30 cm. (Lower part of CCB, under the ground level, can be flooded, though. However, the CCW Pumps are located at the ground level floor, CCB elevation 100.30.) The safety related tanks in the yard are expected to survive at least lower flood levels. (The RWST is also surrounded by the wall.) Furthermore, the plant is expected to be shutdown at the time the plain flooding begins. Time to core uncovery would allow for the implementation of some of alternative ways to achieve heat sink, described in the Appendices to ECA-0.0 and SAMGs, even with NPP plain flooded to certain (lower) level. It may even allow for the flood to recede (depending on the flood maximum). However, the considered flood flow of 7081 m³/s is already fairly above the current estimate of probable maximum flood (6500 m³/s) and is far above the 10000-year flood of 4790 m³/s. Therefore, the flood with flow above 7081 m³/s, such that core damage would be unavoidable is considered to be very low probability event.

Conclusions regarding the margins for external flooding

When NPP plain is to be flooded, it would not happen from the river side, i.e. by water overflowing the dike. Flooding of NPP plain would come from behind. The incoming water would split and the NPP plain would form an island, which would be protected from the river side by the dike. The difference in elevations between the dike and the NPP plain buildings openings is 1.60 m. For example, at flow of 7081 m³/s (worst case flow based on the ANSI/ANS-2.8-1992 scenarios), maximum flow of 136 m³/s would go behind the NPP. From the plant specific study it can be seen that, at this flow, the water level behind the NPP is 154.43, which is by more than 1 m below the NPP plain elevation. It would take much higher flows to actually flood the NPP plain. The most recent studies indicate that actual flooding would not start below some 11000 m³/s. Therefore, the flood with flow such that core damage would be unavoidable is considered to be very low probability event. Having in mind that 1000-year and 10000-year floods are estimated at 4040 m³/s and 4790 m³/s, respectively, it can be expected that the return period for the flood as large as 11000 m³/s would lie in the range of 1E+06 year or larger. From that point cliff edge effect can be defined at this flow which is about 1.7 times higher than existing PMF flow.

Containment integrity

Regarding the margin for containment integrity, the success paths are defined for early releases and for late releases separately. Their characterization can be summarized as follows.

For early releases, one success path is defined: success of containment isolation. This success path applies to all initiator categories considered.

Also, additional success path, which is assumed implicitly, is prevention of core damage.

For late releases, two success paths are defined. The first one is successful operation of reactor containment fan coolers (RCFC). The second one is successful operation of containment spray recirculation in combination with LPSI recirculation through heat exchangers.

As earlier, an additional success path, which is assumed implicitly, is prevention of core damage.

These two success paths are only relevant if early release was avoided. The two success paths apply to all initiator categories considered. In the case of induced total loss of ESW (e.g. due to the clogging of intake structure), the only success path is avoidance of core damage. However, the strategies exist to limit any late release.

As with core damage margin, the margin for the containment integrity function is assessed by evaluating the availability of all success paths following a postulated flooding event, with increasing severity.

The evaluation of containment integrity margin for external flooding starts with the lowest flooding events and continues with gradual increasing of the flooding severity, in terms of a maximum river flow.

Based on the plant specific analyses, the expected containment response would be as follows.

1. River flow approaching 4790 m³/s (10000-year flood).

The NPP plant is expected to be shut down. All NPP SSCs are completely safe. Refer to the discussion above for the core damage margin. Success path for early releases is not challenged. Regarding the late releases, both success paths are considered available.

2. River flow between 4790 m³/s and 7081 m³/s.

At the upper end, the NPP plain is still dry. Nevertheless, a LOOP cannot be excluded due to grid conditions. Also, loss of ESW due to clogging by debris cannot be excluded, although it is not considered likely. In the case of LOOP, all functions for containment heat removal are available. In the case of loss of ESW, the DHR would be implemented by alternative methods (secondary heat sink by AFW TDP, ADP-1.3.030) and means (Appendices to ECA-0.0). The reactor core would be preserved and containment would not be challenged. Success path regarding the early release is not considered to be challenged. Regarding the late releases, in the case of LOOP, both success paths are available. In the case of loss of ESW, described success paths are not available. However, the reactor core would be preserved by alternative methods / means described, and containment would not be challenged.

3. River flow > 7081 m³/s.

As discussed under core damage margin, plant response strongly depends on the maximum water level and duration. With ESW or EDGs lost, the functions required for the containment heat removal would be unavailable. However, reactor core damage, and hence, challenge to containment, can be avoided at water flows significantly higher than 7081 m³/s. Since the plant would be shutdown, the sequence development (to the point of core uncovery) would be slow and would enable the implementation of alternative methods described in the ECA-0.0 Appendices and SAMGs even with NPP plain flooded to a certain level. If the core is preserved, there would be no challenge to the containment. The most recent studies indicate that actual flooding would not start below some 10000 m³/s. For more details, refer to core damage evaluation. For early releases, success path is not considered to be challenged. For late releases: In the case of LOOP with unavailable EDGs, or loss of ESW, both success paths are unavailable. In such a case, containment would not be challenged as long as the reactor core would be preserved by alternative methods / means discussed. Even in the case that core damage (from shutdown state) becomes unavoidable, the development of a sequence to the point of containment challenge would be relatively slow and would enable implementation of some of the strategies for controlling the containment conditions from SAG-6. Those strategies would limit any late releases.

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

Regarding the additional protection against the external flooding from Sava river, several options are currently considered for increasing the elevations of dikes. They are described in the recent plant specific studies.

As noted in the section on seismic margins (section 2.2.1.2), a number of measures have already been implemented at NPP in order to increase the robustness with respect to external events. They include:

- Alternative means to provide suction to AFW pumps or to provide water to SGs directly;
- Alternative means for power supply to CVCS PDP in order to preserve RCS inventory and integrity of RCP seals in induced SBO or Loss of ESW / CCW conditions;
- Alternative means for power supply to selected motor operated valves (MOV), as necessary for the implementation of alternative methods;
- Alternative means for providing water from the external sources to containment;
- Procedures for local operation of AFW TDP and for local depressurization by means of SG PORVs, both without need of DC or instrumentation power;
- Alternative means for makeup of SFP inventory.

It has to be also pointed out that Krško NPP is implementing the upgrade of the existing flood protection by increasing the dikes upstream the Krško NPP (on the left bank of the Sava river and of the tributary Potočnica). This will further increase plant safety.

4 Extreme weather conditions

4.1 Design basis

4.1.1 Reassessment of weather conditions used as design basis

The site of Krško NPP itself lies in the basin of Krško and Brežice which belongs to the region of the western Panonian Boundary Hills, and it is separated from the Panonian Plain by mountains up to 1000 m high (relative height): Gorjanci, Medvednica, Kalnik and Ivančica. The basin is open towards the Panonian Plain through the narrow channels between the above mentioned mountains, along the Sava and Krapina rivers.

The whole region of the site is sheltered from west planetary winds by the Alps and Dinaric Mountains. Therefore, the region is known for having only moderate ventilation, with high frequency of low velocity winds. This is especially valid for the basins within the region, like the basin of Krško and Brežice itself. Strong winds occur mostly during passages of synoptic scale disturbances (front of cyclones) over the region. These synoptic-scale disturbances are strongly influenced and modified when crossing the Alps and Dinaric Mountains.

The climate of the region is continental and partly influenced by the nearby Alps. The influence of the Mediterranean climate is prevented by the Dinaric Mountains. The region is also known for high relative humidity (up to 90% annual averages on location of the AMS tower and up to 80% annual averages on other AMS locations). Frequent occurrence of fog is typical especially in the basins and valleys, with frequent and strong temperature inversions. Severe weather conditions occur quite often, but their severity or intensity is low. Thunder storms winds and heavy precipitation are most frequent in the summer season (March - September). Stagnant atmospheric conditions with high relative humidity and fog are most frequent in the winter season (December - January).

Strong winds in the region occur mostly or together with the most severe thunderstorms. These occur several times per year, (about 7 times per month for the season May - August). The majority of these thunderstorms have no significant effect on the region. The highest instant maximum values of the wind speed are up to 26.5 m/s on ground level stations. The highest half hour average values of the wind speed are up to 13.1 m/s on ground level stations. On 40 m relative altitude AMS tower location instant maximum values recorded are up to 28.3 m/s and highest half hour average values up to 17.3 m/s. On 70 m altitude on AMS tower location instant maximum values up to 26.1 m/s and highest half hour average values up to 22.5 m/s can be expected.

The heaviest 24 hour rainfall recorded was 128.8 mm. The largest rainfall in one month in the same period was 340.7 mm. In the upper part of the Sava river drainage area, precipitation is 50 to 100% higher (NW from Krško).

Typical frequency of hail occurrence is once to twice per year. Ice storms may occur several times each winter. Typical accumulation of ice is approximately 1 – 2 mm; however, icing is more often caused by fog and high atmospheric moisture content than by rain.

The deepest thickness of snow layer recorded in recent decades appeared in 1952 with 76 cm of snow.

Local cyclonic storms have been rarely reported in Slovenia (e.g. destruction of small forest near Postojna, 1964). But there are no data for their occurrence in the considered region.

There are no reported hurricanes in the Krško basin. Tsunamis are not considered as possible because there is no sea or other bigger water body near the plant.

4.2 Evaluation of safety margins

4.2.1 Estimation of safety margin against extreme weather conditions

Structures that house and protect all Safety Class equipment are as follows:

- Shield Building
- Containment Vessel
- Interior Structures
- Auxiliary Building
- Fuel Handling Building
- Intermediate Building
- Control Building
- Diesel Generator Building
- Component Cooling Building
- Essential Services Pumphouse

All listed structures and buildings are designed to withstand severe weather conditions that could occur at NPP Krško.

Intake structures, water systems and chillers

For intake structures and water systems, low and high temperatures represent the challenging point. Same is true for the air cooled equipment. The measures against high and low temperatures are described in following subchapters. For analysis of extreme local rainfall refer to chapter 3.1.1 of this report.

Operation of ESW system (service water – component and emergency cooling) during cold weather season

During the cold weather season when the river water temperature may be near freezing, warm water is diverted from the essential service water to the inlet of the intake structure for de-icing purposes. With the addition of this re-circulated water to the incoming flow of river water, the temperatures of essential service water will increase to 2.22 °C or greater, preventing formation of frazzle ice on the trash rack bars or the traveling screens. Together with additional electrical heating installed on ESW traveling screens and on ESW trash rakes it prevents freezing and loss of function of this part of ESW at extremely low external temperatures of -28.9 °C.

The temperature of the river water is considered to be a maximum 26.7 °C and a minimum of 0.6 °C.

Operation of Chilled Water system (chiller unit) during cold weather season

Each water chiller unit, including its appurtenant electrical components are designed and dedicated for safe and continuous operation under ambient atmospheric pressure and temperatures from -18 to 35 °C. However, in summer time chillers are operable with reduced cooling capacity until extremely high outdoor ambient temperatures of even 48 °C.

During winter when the environmental temperatures go even below -18 °C chillers' operation is still possible.

Heating of equipment

During cold weather, heating is provided for various equipment, pipes and tanks.

Water heating is provided for following water tanks: Refueling Water Storage Tank (RWST), Condensate Storage Tanks (CST), Makeup Water Tank, Demineralized Water Tanks, Pretreatment Water Tanks and Fire Protection Tank.

Heat tracing for various pipes that are thermally isolated but exposed to outside temperatures are constantly switched on during cold months (from 1st of November till 1st of April).

Protection plates are mounted and shades are closed on west wall of turbine building and Steam Header Room.

Surveillance is performed for all open air isolated lines to confirm that isolation is intact to perform its function (around circulating water pumps and circulating water traveling screens, around service water traveling screens, at open air tanks of the condensate system (CY), pretreatment water system (PW), water treatment system (WT), fire protection system (FP), reactor makeup water system (MW), RWST and others).

Also surveillance of cooling liquid is performed for all diesel generators that are exposed to cold air. Day tanks for diesel generators are filled with fuel and heated, so that temperate is above -17 °C

Regular weekly and on extreme low temperatures daily plant surveillance is performed including verification of proper performance of heating steam system and heat tracing system.

Cliff Edge Effects

At lower than design temperatures, some problems may start to arise due to freezing. In this case alternative measures would be initiated. Additional heating would be provided by means of gas heaters located on the site. This action is also applicable in case of heaters failure.

At higher than design temperatures plant would be shutdown to hot standby and if needed further to cold shutdown.

As discussed, exceeding of design temperatures (low or high) would not directly cause a cliff edge effect.

4.2.2 Measures which can be envisaged to increase robustness of the plant

According to above description, no measures were recognized to increase robustness of the plant systems for severe weather risk.

5 Loss of electrical power and loss of ultimate heat sink

5.1 Nuclear power plant

5.1.1 Loss of offsite power

5.1.1.1 Design provisions taking into account this situation, back-up power sources provided and how to implement them

The Krško NPP is one unit plant with one generator rated at 730 MWe. The generator is connected to the two 400 kV switchyard busses via generator load breaker, two step-up transformers 21 kV/400 kV and substation breaker. The 400 kV switchyard is connected to the 400 kV grid with three high voltage transmission lines, one power line to the Elektro-Slovenija network junction Maribor, while the other two (Zagreb I and Zagreb II) are connected to the network of the neighboring country Croatia. The switchyard 400 kV bus is also extended to transformer distribution station (TDS) Krško and connected to 110 kV switchyard via 400 kV/110 kV transformer T411. Two unit transformers T1 and T2 are connected between generator load breaker and step-up transformers and provide normal onsite power supply for two Class 1E (train A safety bus MD1 and train B safety bus MD2) and two non-1E (M1 and M2) 6.3 kV busses.

All four busses can be energized also from station auxiliary transformer 60/30/30 MVA T3 powered through direct underground cable from 110 kV TDS Krško or directly from gas-steam power plant (GPP) Brestanica which is located 7 km from NPP Krško. GPP Brestanica is equipped with three gas powered units of 23 MW capable of blackstart in the event of a breakdown of the 110 kV system and providing electrical power to Krško NPP station auxiliary transformer T3 in 20 minutes.

If transformer and generator protection system is activated, the substation breaker will open, the turbine will trip and will also cause the reactor to trip. The fast transfer of the 6.3 kV busses to the station auxiliary transformer powered from 110 kV TDS Krško will be automatically performed. If this fast transfer is not successful, the undervoltage on each of the safety Class 1E 6.3 kV bus will initiate the bus strip and start of each 3.5 MW emergency diesel generator. At normal frequency and voltage the diesel generators will connect to the safety busses in less than 10 seconds. Safety related equipment will automatically connect to the safety buses per blackout sequence.

If the frequency in the network starts decreasing, then the load increase of the plant is blocked. If frequency is still decreasing, the reactor coolant pumps will be tripped on under-frequency protection and reactor trip will be initiated.

If voltage in the network starts decreasing, the undervoltage protection on each of the Class 1E 6.3 kV safetybus will initiate the bus strip and start of each 3.5 MW emergency diesel generator. At normal frequency and voltage the diesel generators will connect to the emergency busses in less than 10 seconds. Safety related equipment will automatically connect to the emergency busses per blackout sequence.

In the case of a total breakdown of transmission network the system operator is obligated to establish electrical power to the Krško NPP as priority per written agreement. The request for the black start of two or three gas turbines in GPP Brestanica is issued in accordance with instructions. The power to the 110 kV TDS Krško switchyard from GPP Brestanica can be established in 20 minutes. There is also a possibility to establish power directly from GPP Brestanica.

5.1.1.2 Autonomy of the onsite power sources

If offsite power supply is lost, the two 6.3 kV safety buses MD1 and MD2 are powered from their respective 3.5 MW emergency diesel generators. Emergency diesel generators are cooled by air and do not need other systems for cooling. Each diesel has underground reservoir with minimum 85 m^3 of fuel allowing at least 7 days of emergency diesel generator operation.

5.1.1.3 Provisions taken to prolong the time of onsite power supply

The time of operation of emergency diesel generators can be prolonged by stopping one emergency diesel generator. For shutdown of the plant and for maintaining the safe shutdown conditions only one train of safety equipment is needed, one emergency bus and one diesel generator. If one emergency diesel generator is inoperable, then fuel can be transferred from one underground reservoir to another by portable air driven pump.

If additional fuel for emergency diesel generator(s) will be required, then we can use any other diesel fuel available on site. The fuel stored for other alternative diesel generators (20 m^3) and the fuel for auxiliary boilers can be transferred to the underground reservoirs.

5.1.1.4 Measures which can be envisaged to increase robustness of the installation

It is recognized that the core damage frequency for events initiated by loss of off-site power can be reduced by installation of third emergency diesel generator. This modification is now in installation phase which is planned to be finished in 2012. The third emergency diesel generator will be located in a separate building with the third safety bus which could be connected to either one of the existing safety buses.

5.1.2 Loss of offsite power and of onsite back-up power sources

5.1.2.1 Loss of offsite power and loss of the ordinary back-up source

5.1.2.1.1 Battery capacity and duration

125 V DC system consists of two independent and redundant trains. Each train consists of a full capacity 125 V DC lead-acid 60 cell battery, 125 V DC switchboard, solid state battery charger and required distribution boards. The battery chargers supply power to the DC system and provide float charge to the batteries during normal operation. Upon loss of station AC power, the entire DC load is supplied by the batteries. The important instrumentation and control is powered from 118 V distribution which is powered through inverters connected to 125 V DC system.

Each 125 V battery charger is sized to carry normal plant operation DC loads of one train while recharging a fully discharged battery in 12 hours. Each train has access to an installed swing charger which in turn can be fed from its associated train 400 V AC source. Interlocks are provided to ensure separation of the redundant trains.

The batteries are sized to supply DC loads as defined above for a minimum of four hours with a final discharge of 108 V (1.80 V per cell). The batteries have sufficient capacity per design to cope with a 4 hour station blackout (loss of all AC power), to provide safe shutdown of the unit. The capacity of each 125 V battery is 2080 Ah.

Establishing alternative power supply to the bus LD11 and to battery chargers from one of the two alternative portable diesel generators (600 kVA or 1000 kVA) will assure the long time availability of DC batteries and of 118 V AC instrumentation power supply (up to 72 hours since fuel is stored at the plant for this time period, or even longer if fuel would be supplied form outside of the plant).

Emergency operating procedures instruct the operators to disconnect all non-essential DC loads. Based on plant specific best estimate DC study and with the actions of the operating crew to disconnect all non-essential DC loads, the above mentioned 4 hours will be extended to and above 16 hours for train A and 13.5 hours for train B. However with the multiplication of additional portable diesel generators, the instruction to strip all non-essential DC loads loses priority as the diesel generators ensure much longer availability of essential instrumentation.

The plant is further equipped with a non safety grade 220 V DC system sized to provide power to Turbine Emergency Oil Pump, Emergency Seal Oil Pump, Emergency Lighting Distribution Panel, DC panel for control of 400 kV substation breakers and Inverters for Process Information System in the event of AC power failure. The 220 V battery is sized to supply the above mentioned DC loads for a minimum of four hours. The battery has a final discharge voltage of 1.80 V per cell. The capacity of the battery is 2175 Ah. The 220 V charger can be powered through transfer switch from non-safety MCC212 or safety MCCD124, which can be powered from 400 V safety bus LD11.

5.1.2.1.2 Autonomy of the site before fuel damage

As described in station blackout analysis, the Krško NPP plant is in a category of plants with four-hour design autonomy. The plant is designed to maintain safe shutdown conditions for four hours in case of loss of both offsite and onsite power:

- In seismically qualified condensate storage tanks there is enough water for removing the decay heat through both steam generators.
- Additional nitrogen gas is provided to auxiliary feedwater control valves for filling both steam generators with turbine driven auxiliary feedwater pump and to steam generator power operated relief valves for releasing steam from steam generators.
- Safety batteries capacity ensure power to 118 V instrument power supply.
- Opening doors ensure appropriate temperature in turbine driven auxiliary feedwater pump room and in main control room cabinets.
- Containment isolation can be done with locally closing isolation valves.
- With local actions to isolate letdown line the inventory loss is minimized.

Safe plant condition can be prolonged or assured in the case of malfunction of any installed equipment by the use of the SAME equipment. This equipment can be connected to the various systems through installed connection points by equipment operators from shift crew and fireman form fire team on site in less than one hour. Regular trainings and drills for shift crews and fire team are conducted periodically.

It has been determined by T/H analysis of the worst case scenarios considering unavailability of the AFW TDP that if temporary SGs injection equipment can be deployed and made functional in 1 hour, enough margin exists to prevent core damage.

As a consequence of a loss of seal injection and loss of thermal barrier cooling the reactor coolant pump seals are exposed to high temperature of the primary coolant. This could cause seal degradation and seal leakage flow up to 21 gpm and loss of the primary coolant inventory. If nothing is done, this will lead to core uncovery due to inventory loss, fuel damage and consequently to primary system failure. Operator actions are required to cool down and depressurize the primary system with steam generators. This action will decrease seal leakage flow and allow passive injection of additional borated water from accumulators and gaining additional time for electrical power restoration.

Emergency operating procedures provide actions to mitigate deterioration of reactor coolant system conditions while AC emergency power is not available. The following major actions are defined:

a. Maintaining auxiliary feedwater flow to both steam generators with turbine driven auxiliary feedwater pump which can be controlled from control room or locally.

   This action ensures decay heat removal. The flow control valves are air-operated and are provided with a 4-hour supply of nitrogen gas. Water sources are two condensate storage tanks. Each has a capacity of approximately 757 m^3. The capacity of each condensate storage tank is based on operation of the system for two hours at hot shutdown followed by approximately four hours of cooldown. The low level reserved for the auxiliary feedwater pumps is equivalent to approximately 860 m^3 (in both tanks together).

If condensate storage tanks' levels are decreasing, then operators will try to establish filling condensate storage tanks from demineralized water tanks. The power to the demineralized water transfer pump can be established from the diesel generator in switchyard if available. The diesel generator in the switchyard can also provide power to the communication system and to 220 V charger which provides power to 220 V battery and important 220 V DC loads like emergency lighting and inverters for process information system (PIS).

The alternative methods for filling condensate storage tanks can be established with portable fire pumps which can provide water from demineralized water tanks, fire protection tank, main condensers, circulating water tunnel or from Sava river.

If condensate storage tanks are not available, then water flow can be established directly to the suction of AFW TDP by the same alternative methods.

b. Minimizing the reactor coolant system inventory loss by isolating letdown lines.

The guidance is provided for establishing alternative power from 150 kVA portable diesel generators to the associated motor control center for closing at least one letdown isolation valve. If both letdown isolation valves in conjunction with the letdown orifice isolation valves remain open, a leak path from reactor coolant system to the pressurizer relief tank via the letdown line relief valve exists.

c. Restoring power to any AC emergency bus by starting at least one emergency diesel generator or by establishing off-site power supply.

If power supply cannot be established, operators can establish power to the 400 V bus LD11 from one of the two available alternative diesel generators. The current/power on cables from these diesel generators is limited to 574 A / 400 kVA.

From LD11 the power can be established to the lighting distribution panels to establish half of normal lighting and complete emergency lighting providing power to 220 V charger. Another important load powered from LD11 is positive displacement pump (PDP) which doesn't need cooling and can provide charging flow from refueling water storage tank and/or boric acid tanks to the reactor coolant system. This charging flow will compensate inventory losses in reactor coolant system and with borating reactor coolant system the re-criticality during cooldown will be prevented.

By energizing 400 V safety bus LD11 operators can establish power to the battery charger A which can provide power to the DC distribution panel train A and to charge the 125 V batteries A and through inverters 1 and 3 the instrumentation distribution panels 1 and 3. DC distribution panel train B can be powered through swing charger A-B. This method is described in system operating procedure and is used in regular outage as temporary modification.

Power to the instrumentation distribution panels 2 and 4 can be established also by energizing motor control center MCCD211 with one of the three 150 kVA portable diesel generators. Power to instrumentation busses 2 and 4 is provided through regulating transformer EE115XFRJ301. If power to 118 V instrumentation system cannot be established or in case of loss of control room, operators can establish alternative power to the shutdown panel train A with two 220 V petrol driven generators with transformation to 118 V to establish power for essential instrumentation.

d. Depressurizing reactor coolant system by depressurizing the steam generators.

This action will minimize RCS leakage, prevent deterioration of RCP seals, permit makeup of water from SI accumulators to the reactor coolant system and also permit filling of steam generators in the case of inoperable turbine driven auxiliary feedwater pump. It is noted that the primary coolant loss is rather low and rapidly decreased with RCS cooldown and depressurization. RCP seals are also cooled by cooling of the RCS. Consequently the core does not uncover within 7 days, therefore, the core, reactor coolant system and the containment integrity are not endangered. Depressurizing the steam generators is performed by opening steam generator power-operated relief valves (SG PORVs). SG PORVs have nitrogen gas for four hours of PORV operations. If SG PORVs cannot be operated, then instructions are given to locally open SG PORVs using compressed air from portable diesel compressor and local pressure regulators or manually.

As alternative to the SG PORVs the main steam safety valves can be used for depressurization of steam generators.

Depressurizing steam generators to 1.47 MPa will allow filling water in both steam generators with high pressure portable FP pump on fire track through auxiliary feedwater system, main feedwater system, blowdown system or condensate system in the case of inoperable AFW TDP. Fire track reservoir can be filled with water from demineralized water tanks, fire protection tank, from main condensers, circulating water tunnel or from Sava river.

If high pressure portable FP pump cannot be used, operators are instructed to isolate SI accumulators using power from portable 150 kVA diesel generators connected to the associated motor control centers. This action will prevent injection of nitrogen from accumulators into the reactor coolant system which could inhibit natural circulation cooling of reactor using the steam generators. When SI accumulators are isolated, then the steam generators can be depressurized to 0.78 MPa. This action will allow filling of both steam generators with normal pressure portable fire pump with water from demineralized water tanks, fire protection tank, from main condensers, circulating water tunnel or from Sava river.

e. To enhance the equipment cooling, certain doors are open:
   - doors of main control room cabinets,
   - door of turbine driven auxiliary feedwater pump (AFW TDP),
   - doors between intermediate building and turbine building.
f. In case of inadequate spent fuel cooling the operators are instructed to initiate spent fuel pool makeup using alternative equipment as described in section 5.2.2 of this report.
g. No cliff edge effects have been identified for period more than 7 days because usage of alternative equipment assures reactor coolant inventory control and decay heat removal.
h. If secondary heat sink is lost and cannot be established in timely manner to prevent core damage, the technical support center can decide to start gravity drain flow from the refueling water storage tank to the containment sump.

   This action is based on severe accident management guidelines strategies described in section 6.2.1 of this report. The basis for starting this action when still performing emergency procedures is establishing gravity drain when containment pressure still allows this. Injecting borated water into containment can delay and possibly prevent vessel failure and will delay containment failure when core damage occurs.

   The gravity drain flow from the refueling water storage tank to the containment sump can be established by using power from portable 150 kVA diesel generators connected to the associated motor control centers to open the valves on emergency core cooling recirculation lines in protective chambers.

Onsite alternative equipment such as diesel generators, portable diesel aggregates, portable petrol driven generators, portable petrol or diesel driven fire pumps and diesel compressors are used to prevent fuel degradation as described in this section. If onsite sources for diesel or petrol fuel are not enough for long-term operation of this alternative equipment, the external delivery will be necessary.

5.1.2.1.3 (External) actions foreseen to prevent fuel degradation

External support from outside organizations is not expected and is not necessary in early phase of the event (first 72 hours).

5.1.2.1.4 Measures which can be envisaged to increase robustness of the installation

- Two petrol driven 125 V aggregates will be available to provide the power to DC system panels in case of loss of DC main distribution panels A and B.
- Two high pressure mobile fire protection pumps will be available for possibility to remove decay heat in early stage after reactor shutdown and depressurizing steam generators.

5.1.2.2 Loss of offsite power and loss of the ordinary back-up source, and loss of any other diverse back-up source

Krško NPP does not have any other diverse back-up sources of power supply.

5.1.3 Loss of the ultimate heat sink

5.1.3.1 Design provisions to prevent the loss of the ultimate heat sink

The essential service water system (ESW) provides cooling water to the component cooling system (CCW) and boron thermal regeneration system (BTR) to transfer the plant heat loads from these systems to the ultimate heat sink (UHS), the Sava river. The system also serves as a backup safety-related source of water for the auxiliary feedwater system (AFW).

The CCW provides cooling for safety systems and engineered safety feature systems. The system operates during all plant operational phases performing normal plant functions as well as safety functions. Therefore, the ESW is a safety-related system with isolation capabilities and redundancy in components and features necessary to satisfy the requirements of these systems under normal or accident conditions combined with a loss of offsite power and any single active or passive failure.

ESW operates during any plant normal or accident condition and during a safe shutdown earthquake with the loss of offsite electric power and any single failure, thus satisfying the safety function and single failure criterion required for this system. The ESW is classified as a Safety Class 3 and Seismic Category I system.

A low dam across the Sava river is used to maintain the water level at a nominal elevation of 150.00 m.a.A.s.l. However, the ESW is designed for operation with any water level varying from the original minimum river level, at an elevation of 147.85 m.a.A.s.l., to a maximum flood level at an elevation of 156.60 m.a.A.s.l. The temperature of the river water is considered to be a maximum of 26.7 °C and a minimum of 0.6 °C.

Dam threshold is designed to continue with its function in case of SSE and to form a pool of capacity 450000 m^3 from which ESW is provided by water together with the bank river protection with the same design basis. The loss of upstream Sava river flow will not disturb ESW cooling function (water flow of 1.06 m^3/s). The highest water temperature rise in this pool, in the case of loss of river flow, is expected to be expected 8.1 °C with the very conservative assumption of no heat transfer from the pool during the 30 day cooldown.

The potential for freezing in the intake structure and piping is considered, with necessary design features included to provide freeze protection.

Appropriate instrumentation is provided in the control room to indicate the status of the system during normal and accident conditions.

The ESW is an open loop cooling system in which water from an intake structure on the Sava river is pumped through the heat exchangers or components to be cooled and returned to the river through a discharge structure.

The system basically consists of two independent loops. Each loop provides cooling water to one component cooling heat exchanger. Cross connections allow the boron thermal regeneration chiller to be supplied with cooling water from either loop. The cross connections are furnished with valves to maintain isolation between the loops. A branch line is provided from each loop through which river water can be delivered as a backup water supply to the auxiliary feedwater system.

Three ESW pumps are provided. One pump is connected to each loop. The third pump is a spare which can be aligned to either loop through cross connections to maintain operability of both loops while one pump is out of operation for maintenance purposes. Valves are provided in the cross connections for maintaining isolation between the loops.

One safety class, automatic, self-cleaning strainer is installed in the discharge line of each pump. Each strainer is equipped with a safety class backwash line for debris removal. The strainer and backwash line are in operation whenever the associated ESW pump is operating. The backwash line contains an orifice to limit backwash flow between 5% and 10% of strainer flow.

The intake structure incorporates redundant inlets and channels for supplying river water to the pumps. Each channel includes a trash rack with a movable trash rake followed by a traveling water screen. The two channels are connected through two 0.91 m openings located on the inlet side of the trash racks and discharge to a common forebay serving the three pumps. The inlets, channels and traveling screens are adequately sized for handling the flow for both loops (plus the fire service water pumps, which are also located in the intake structure). Thus, the intake structure provides complete redundancy in flow paths to the ESW pumps.

Water for the traveling screen wash systems is supplied from the ESW loops. Water is supplied through self-cleaning strainers to horizontal, centrifugal pumps which then deliver it to the traveling screen spray systems. The branch lines of the loop are joined by a normally open connecting line; thus, water is supplied to both screen wash circuits when only one ESW loop is in operation. Valves in the connecting line close automatically in response to a safety injection signal, or in the event of a loss of offsite power, to establish isolation between the loops. Each screen wash system then receives its water supply from its respective loop. Normally, closed valves in pump discharge lines prevent flow through the systems when the screen wash systems are not operating.

A non-safety class ball cleaning system is installed on the ESW side of the component cooling heat exchangers. Four normally closed safety-related isolation valves provide isolation between the ESW loops and between safety and non-safety related piping. Procedural controls ensure that only one component cooling heat exchanger is cleaned at a time.

All electrical equipment in an ESW loop, including screen wash system, associated traveling screen and one trash rake, and strainer is supplied from the same emergency bus. Electrical and physical separation is maintained between the buses supplying the two loops. The buses are normally supplied by offsite power sources. In the event of a loss of offsite power, each bus is supplied by separate, on site emergency diesel generators.

The ESW loops discharge into separate compartments in the discharge structure. The water is then released to the river over weirs located at the maximum flood elevation of 156.50 m.a.A.s.l. The separation of the loops through the discharge structure and the level of the weirs eliminate any potential for uncontrollable flooding through a break in the ESW piping.

A portion of the ESW discharge is recirculated when the temperature of the river water is near freezing to prevent the formation of frazil ice within the intake structure. The deicing flow is returned to the intake structure through lines from the discharge structure. A line is provided from each compartment of the discharge structure. Each line discharges through a split header into both channels, upstream of the trash rack. The deicing lines are designed to provide a recirculation flow of up to 680 m^3/h from both operating loops. The recirculation flow is controlled by a manual throttle valve to maintain the temperature of the water flowing through the intake structure above 2.22 °C.

The operation of the ESW loops is initiated either automatically or by manual operator action. A loop becomes operational with the startup of the pump aligned with that loop. One loop will be in operation at all times. Both loops will be operated during a normal plant startup or shutdown, or at any time the plant cooling requirements necessitate the operation of both component cooling heat exchangers. The second loop is started automatically following a loss of offsite power or an accident resulting in a safety injection signal.

During normal plant operation the second loop provides backup to the operating loop. The pump in this loop starts automatically in response to a low pressure signal if a failure of the pump in the operating loop occurs. However, as the backup loop only delivers water to the alternative component cooling loop (heat exchanger) the operation of this system must be switched to these loops in order to continue heat removal from the system with the ESW. Correspondingly, if a failure occurs in the component cooling operating loop and operation is switched to the alternative loop, the ESW backup loop must be started to provide cooling water to the new operating loop.

Normal operation of the traveling water screens and the screen wash systems is initiated automatically by bubbler systems across the screens. Each screen (and its wash system) is controlled by a separate bubbler system. The bubbler starts the screen wash pump; however, interlocks prevent the pump from starting if the pressure in the supply line is low (indicating that the suction line is not open to an operating loop) or if the valve in the pump discharge line has not started to open. Operation of the screen is initiated when the pressure in the screen wash header is adequate to ensure proper cleaning of the screen. Operation is terminated by a timer subsequent to elimination of the high differential level across the bubbler.

Operation of a traveling screen can also be initiated by a manual start of the screen wash pump from a local panel. The pump is also started automatically if there is a loss in air pressure to the bubbler system.

The ESW strainers are interlocked with the associated pump so that the strainer motor starts and the backwash valve opens on an ESW pump start. The strainers continue to operate and the backwash valve stays open as long as the associated ESW pump is operating.

Sufficient instrumentation and alarms are provided for monitoring, evaluating and controlling the operation of the ESW.

The ESW piping between the intake structure and the component cooling building, the component cooling building and discharge structure, and the discharge structure and intake structure is routed underground. For long term corrosion protection, the underground piping is protected by a wrapped, coal tar enamel coating, and cathodic protection is provided where soil samples indicate such action is required. The pipe is designed and provided with slip joints as necessary to provide the capability for rotation and extension to preclude damage due to soil settlements, seismic events, or differential structure and soil movement. The piping is buried 2 meters below the plant grade elevation of 155.20 m.a.A.s.l. This locates the pipe below the frost line and provides sufficient earth coverage for protection from surface loads.

5.1.3.2 Loss of primary heat sink (access to water from the river or the sea)

In scenario with loss of heat sink there is assumption that the loss means the connection between the pumps and loads is lost. All other systems operate normally and water is available from the Sava river.

If the flow from ESW to the component cooling heat exchangers cannot be established, operators according to procedures take actions to shutdown the plant and perform compensatory measures. Main actions which need to be performed are:

- Stop both reactor coolant pumps

- Establishing of reactor coolant pump's seal injection flow with positive displacement pump
- Stabilization of reactor coolant temperature
- Establish boration flow to the reactor coolant
- Establishing feed flow to both steam generators with turbine driven auxiliary feedwater pump or one main feedwater pump
- Checking pressurizer level control
- Isolation of letdown flow
- Checking pressurizer pressure control
- Establishing instrument air with condensate polishing compressor (cooled with pretreatment water)
- Verifying that reactor coolant is cooled by natural circulation
- Maintaining stable plant condition with control of feed flow to both steam generators and with steam release into the atmosphere or condenser if available
- Makeup of condensate storage tanks from available water sources
- Actions to restore the essential service water
- Natural circulation cooldown.

5.1.3.2.1 Autonomy of the site before fuel damage

»No load« temperature is maintained with natural circulation of reactor coolant, feeding both steam generators (SG) with feedwater and steaming with SG PORVs. Auxiliary feedwater turbine driven pump (AFW TDP) normally provides feed flow to both SGs, as an alternative main feedwater pumps can also be used. SG PORVs are used to remove decay heat from the core, by steaming in to the atmosphere; steam dump can be used if condenser is available.

Letdown flow has to be isolated due to loss of cooling. This will cause that level in the pressurizer will start to increase. Charging flow which is provided with positive displacement pump (PDP) has to be set to minimum.

Inoperability of ESW for longer period will require the plant cooldown to Hot Shutdown mode. Cooldown rate will be adjusted based on level in pressurizer which is maintained with charging flow into the seal injection lines and water contraction. Cooldown should be performed in steps to verify that the level in pressurizer can be maintained. Temperature will be controlled below 177 °C and it is recommended to maintain reactor coolant temperature between 130-150 °C. This will ensure enough steam pressure to run AFW TDP and also enough steam pressure to remove decay heat from the core with releasing steam from SGs.

Reactor coolant temperature and pressure will be maintained at that level until cooling of the component cooling heat exchanger is restored.

Water source for long-term operation is provided with design provision of demineralized water system or pretreatment water system. Alternative water sources can be used for filling the condensate storage tanks (CST): demineralized water storage tanks, fire protection tank, condenser, circulating water tunnel, Sava river, and potable water from city of Krško.

In case of inoperable CST and operable AFW pumps, water can be delivered to the suction of AFW pumps with portable fire protection pumps from the following water sources: demineralized water storage tanks, fire protection tank, condenser, circulating water tunnel, Sava river.

In case AFW TDP is not available portable fire protection pumps can be used to supply water into both SGs. These pumps have enough capacity to remove the decay heat from the core and to maintain the level in both SGs to provide natural circulation on primary side.

Increased water level in the pressurizer can be compensated with decreasing injection flow to the RCP with adjustment of the seal injection flow to the minimum value of 1.6 m^3/h per RCP. After a long period of time the level in the pressurizer will be increased, temperature on primary side will be maintained between 130 and 150 °C and pressure will be around 1.96-2.45 MPa. To prevent solid operation it will be necessary to start decreasing level in pressurizer. Low pressure letdown flow cannot be used due to inoperable RHR system (loss of component cooling). Level in pressurizer can be decreased with opening pressurizer PORVs and leaking RCS coolant into the pressurizer relief tank. Also, normal letdown flow can be used in case of bypass of CVCS demineralizers.

Source of water for PDP pump is refueling water storage tank (RWST).

No cliff edge effects have been identified for period more than seven days because usage of alternative equipment assures reactor coolant inventory control and decay heat removal. All alternative equipment can be connected to the various systems through installed connection points by equipment operators from shift crew and firemen from fire team on site in less than one hour. Regular trainings and drills for shift crews and fire team are conducted periodically.

5.1.3.2.2 (External) actions foreseen to prevent fuel degradation

External support from outside organization is not expected and is not necessary in early phase of the event (first 72 hours). Equipment stored on site will be used. All necessary actions can be performed by shift crew and additional personnel from Technical Support Center (TSC) and Operating Support Center (OSC). Organization for emergency situation is established according to the national Radiological Emergency Response Plan (RERP) and Emergency Implementation Procedures (EIP). Plant well water or potable water from Krško city can be used for different purposes such as filling the CSTs, reactor water storage tank, filling the SFP, boric acid tanks, reactor makeup water storage tank and for preparation of additional demineralized water.

Based on situation and cause of the event there may be a need for support from outside organization with the heavy equipment to participate in restoration of the heat sink.

5.1.3.2.3 Measures which can be envisaged to increase robustness of the installation

Two high pressure mobile fire protection pumps will be purchased for possibility to remove decay heat in early stage after reactor shutdown and depressuring steam generators.

Operation of centrifugal charging pump and high pressure safety injection pump (HPSI) is important to provide enough flow to have capability to maintain level and pressure under control. Connection provisions to establish cooling water for the oil coolers of charging and safety injection pumps from alternative sources would assure pump operation without normal cooling.

Installation of the connection on the non-safety related part of the piping and connection with portable/mobile fire protection pump can provide alternative cooling for component cooling heat exchanger.

The connection is standard type A for fire protection equipment and it can be easily used with crew onsite. Source of water for cooling can be water from the Sava river or any other water which is available onsite. This alternative cooling has a limited cooling capacity, but it will have enough capacity to allow operation of the centrifugal charging pump, HPSI pump and even motor driven AFW pump or any other small heat load which will be necessary. Control room is cooled by the chilled water system, which is independent from the CCW and ESW.

Alternative cooling can also be established with installation of 8'' tee on the existing 24'' ESW line to CCW heat exchangers to provide alternative connection for fire protection pump with higher capacity and connection size of 8''. The capacity of already ordered pump »HFS HydroSub 450 floating unit«, is 720 m^3/h, which can provide enough heat removal for one train of ECCS and also to remove decay heat from SFP.

Alternative residual heat removal could be established with skid mounted pump and heat exchanger and connection points to RHR system.

New water line from the Krško HPP, which will be installed in the near future, will provide alternative way of cooling the CCW heat exchanger. Gravity force will be used as passive cooling system.

5.1.3.3 Loss of »primary« heat sink and »alternative heat sink«

Krško NPP does not have an »alternative heat sink«.

5.1.4 Loss of the primary heat sink, combined with station blackout

5.1.4.1 Autonomy of the site before fuel damage

Loss of the primary heat sink, combined with station blackout reduce the capability to use the existing as designed safety equipment, which need to have electrical power supply and need to be cooled.

To fulfill the requirements of each safety function, equipment which is onsite can be used.

Decay heat removal is achieved with AFW TDP and steam relief into the atmosphere through SG PORVs. For the first 4 hours (or more as per 5.1.2.1.1), there are batteries and compressed nitrogen in bottles to operate valves (SG PORVs and control valves for AFW TDP). During that period alternative source of power and compressed air can be established or we can manually control the speed of AFW TDP and manually release the steam from SGs to control the decay heat removal. If SG PORVs cannot be operated, instructions are given to locally open SG PORVs using compressed air from portable diesel compressor and local pressure regulators or manually. As an alternative to the SG PORVs the main steam safety valves can be used for depressurization of SGs.

For the alternative power supply we can use one of two alternative diesel generators, which are located on the highest ground elevation which is safe in case of flooding and provide electrical power to the 400 V bus LD11.The current/power on cables from these diesel generators is limited to the 574 A/400 kVA.

From LD11 the power can be established to the lighting distribution panels to establish half of normal lighting and complete emergency lighting providing power to 220 V DC charger.

Another important load powered from LD11 is the PDP which doesn't need cooling and can provide charging flow from RWST and/or boric acid tanks to the RCS. This charging flow will compensate inventory losses in RCS and with borating reactor coolant system the recriticality during cooldown will be prevented.

Operators can establish power from LD11 via motor control center MCCD111 to the battery charger A which can provide power to the DC distribution panel train A and to charge the 125 V batteries A and through inverters 1 and 3 the instrumentation distribution panels 1 and 3. DC distribution panel train B can be powered through swing charger A-B. This method is described in system operating procedure and is used in regular outage as temporary modification.

Power to the instrumentation distribution panels 2 and 4 can be established also by energizing motor control center MCCD211 with one of the three 150 kVA portable diesel generators. Power to instrumentation busses 2 and 4 is provided through regulating transformer EE115XFRJ301.

If power to 118 V instrumentation system cannot be established or in case of loss of control room, operators can establish alternative power to the shutdown panel train A with two 220 V petrol driven generators with transformation to 118 V to establish power for essential instrumentation.

As described above, decay heat removal is independent of heat sink and component cooling media. Stabilization above 130 °C is recommended to have enough driving steam to run the AFW TDP. At this operating condition the plant can stay as long as there is enough water to remove decay heat from primary side and to protect the integrity of the nuclear fuel provided RCS leak is minimized and controlled by depressurization/injection as explained before.

Filling of CSTs can be accomplished with demineralized water transfer pumps or with alternative methods. The power to the demineralized water transfer pumps can be established from the diesel generator in switchyard. The diesel generator in the switchyard can also provide power to the communication system and to 220 V charger which provides power to 220 V battery and important 220 V DC loads like emergency lighting and inverters for process information system.

Alternative water sources can be used for filling the CSTs: demineralized water storage tanks, fire protection tank, condenser, circulating water tunnel, Sava river, and potable water from city of Krško.

In case of inoperable CSTs and operable AFW pumps, water can be delivered to the suction of AFW pumps with portable fire protection pumps from the following water sources: demineralized water storage tanks, fire protection tank, condenser, circulating water tunnel, Sava river.

All alternative portable and mobile equipment is located onsite at least 100 m away from the reactor as required by NRC regulation for alternative equipment. Equipment is powered from diesel or gasoline engines, with enough storage capacity for 72 hours at rated load.

Krško NPP can be in this condition for at least 7 days provided that delivery of additional fuel for alternative equipment is assured after 72 hours.

In case AFW TDP is not available portable fire protection pumps can be used to supply water into both SGs. These pumps have enough capacity to remove the decay heat from the core and to maintain the level in both SGs to provide natural circulation on primary side.

No cliff edge effects have been identified for period more than 7 days because usage of alternative equipment assures reactor coolant inventory control and decay heat removal. All alternative equipment can be connected to the various systems through installed connection points by equipment operators from shift crew and firemen from fire team on site in less than one hour. Regular trainings and drills for shift crews and fire team are conducted periodically.

5.1.4.2 (External) actions foreseen to prevent fuel degradation

External support from an outside organization is not expected and not necessary in the early phase of the event (first 72 hours). Equipment stored onsite will be used. All necessary actions can be performed with shift crew and additional personnel from TSC and OSC. Organization for emergency situation is established according to the RERP and EIP procedures.

For long-term operation external support is needed for diesel and gasoline supply to run the portable alternative equipment.

Water sources are also important for long-term operation. Primary water sources are CSTs, demineralized water storage tanks, potable water, well water and also the Sava river. External support can also be provided by enough water capacity from Krško potable water source or any other available water source.

5.1.4.3 Measures which can be envisaged to increase robustness of the installation

- Two high pressure mobile fire protection pumps will be purchased for possibility to remove decay heat in early stage after reactor shutdown and depressuring steam generators.
- Mobile water station with enough capacity for all needs for decay heat removal.

- Third independent diesel generator with safety bus, which can be connected to both existing safety buses.
- Provision to connect mobile diesel generator of capacity 2000 kVA to switch gear of the third diesel generator.
- Connection point for alternative cooling of component cooling heat exchanger.
- New water line from Krško HPP could provide alternative way of cooling the component cooling heat exchanger by a passive means.
- Alternative residual heat removal could be established with skid mounted pump and heat exchanger and connection points to RHR system.

5.2 Spent fuel pool

5.2.1 Loss of offsite power

5.2.1.1 Design provisions taking into account this situation, back-up power sources provided and how to implement them

The spent fuel pool (SFP) cooling system consists of two 100% pumps, three heat exchangers, one mixed bed demineralizer, one 5-micron filter and associated piping and valves. During normal operation of the SFP cooling system water is drawn from the SFP by one SFP pump and is pumped through the tube side of one heat exchanger, and returned to the SFP. Each suction line, which is protected by a strainer, is located at the elevation four feet below the normal water level, while the return line terminates at the elevation six feet above the top of the fuel assemblies, and contains an anti-siphon hole near the surface of the water to prevent gravity drainage. The system is controlled locally.

If necessary to remove the complete core from the reactor (121 fuel assemblies) with the racks full from previous refuelings (1573 fuel assemblies) thus filling the racks to capacity (1694 fuel assemblies in total), the SFP cooling system is capable of maintaining the SFP water temperature at or below 72.4 °C when the SFP heat exchanger SFAHSF02 is supplied with component cooling water at the design flow and temperature. The use of the heat exchanger SFAHSF01 together with SFAHSF 03 would result in a higher temperature of the spent fuel cooling water, i.e. 73.5 °C.

SFP cooling pumps are powered from safety related 400 V busses LD11 and LD12. In the event of a loss of offsite power the safety-related emergency buses can be powered either from 110 kV switchyard TDS Krško through station auxiliary transformer or from emergency diesel generators as described in section 5.1.1.

If diesel generators are started and blackout or safety injection sequence is initiated, the breaker for operating SFP cooling pump will open. One SFP pump will be then manually started as instructed by procedures.

5.2.1.2 Autonomy of the onsite power sources

The autonomy of the emergency diesel generators is described in section 5.1.1.2.

5.2.1.3 Provisions taken to prolong the time of on-site power supply

Provisions taken to prolong the time of emergency diesel generators operation are described in section 5.1.1.3.

5.2.1.4 Measures which can be envisaged to increase robustness of the installation

Measures which can be envisaged to increase robustness of the installation are described in section 5.1.1.4.

5.2.2 Loss of offsite power and of onsite back-up power sources

5.2.2.1 Loss of offsite power and loss of the ordinary back-up source

5.2.2.1.1 Battery capacity and duration

The battery capacity and duration is described in section 5.1.2.1.1.

SFP temperature and level instrumentation is normally powered from the process information system. In case of loss of normal power, the dedicated battery is provided to power SFP temperature and level instrumentation.

5.2.2.1.2 Autonomy of the site before severe accident

If loss of all AC power occurs, SFP cooling pumps would be lost and the cooling flow to the SFP heat exchangers would be lost. The temperature of water in SFP would start to increase. At that point heat removal from the SFP is established by water boiling in the SFP. For maintaining the constant water level in the SFP it is required to deliver adequate water flow. In the case of maximal possible decay heat value 8.5 MW from the fuel in the SFP time to boil at minimum allowable level is 4 hours and 28 minutes and 14.1 m^3/h of makeup water flow to the SFP is required to maintain constant water level after start of boiling. If water is not delivered into the SFP, then the USAR limit 3.05 m above of the top of fuel elements would be reached after 47 hours after event initiation.

Procedures instruct the operators to monitor the SFP level and temperature and to initiate the makeup to the SFP.

If the power to the bus LD11 is established as described in section 5.1.2.1.2, then the normal makeup to the SFP can be established from RWST through purification lines or from reactor makeup storage tank.

Alternative means for establishing spent fuel pool makeup:

- Pumping water from water pretreatment tanks with portable fire pump to the system for purification of SFP water surface.
- Providing water from fire protection hydrant network to the system for purification of SFP water surface. This method requires pressurized fire protection hydrant network by installed diesel fire pump.
- Pumping water from pool near water pretreatment building with portable fire pump to the system for purification of SFP water surface.
- Pumping water from circulating water intake pool with submersible fire pump and fire track to the system for purification of SFP water surface.
- Pumping water directly to SFP from fire protection system.

If water in the SFP is decreasing even if makeup to the SFP is established, then operators are instructed to establish water spray over the spent fuel before the water drops below 3.05 m above the spent fuel elements. Water spray with portable fire protection nozzles ensure adequate cooling of spent fuel elements. The priority of water sources is prescribed as follows: fire protection hydrant network, water pretreatment tanks, pool near water pretreatment building, circulating water intake and circulating water outlet pool.

No cliff edge effects have been identified for period more than seven days because usage of alternative equipment assures spent fuel heat removal. All alternative equipment can be connected to the various systems through installed connection points by equipment operators from shift crew and firemen from fire team on site in less than one hour. Regular trainings and drills for shift crews and fire team are conducted periodically.

5.2.2.1.3 (External) action foreseen to prevent fuel degradation

External support from outside organizations is not necessary and is not expected in early phase of the event (first 72 hours).

5.2.2.1.4 Measures which can be envisaged to increase robustness of the installation

- Installation of fixed piping around the SFP with spray nozzles and connections for portable fire pumps.
- Skid mounted pump and heat exchanger for alternative cooling of SFP.

5.2.2.2 Loss of offsite power and loss of the ordinary back-up source, and loss of any other diverse back-up source

Krško NPP does not have diverse back-up sources of power.

5.2.3 Loss of the primary heat sink

5.2.3.1 Design provisional autonomy of the site before severe accident

The Krško spent fuel storage area design is in compliance with NRC Regulatory guide 1.13.

The Spent Fuel Pool Cooling and Cleanup System (SFPCCS) is designed to remove the decay heat generated by spent fuel assemblies stored in the SFP. A second function of the system is to maintain clarity (visual) and purity of the spent fuel cooling water and the refueling water.

The SFPCCS is designed to remove the decay heat produced by 1694 spent fuel assemblies in storage following the refueling of 40% of a core (48 fuel assemblies) plus a conservatively large number of spent fuel assemblies from previous refuelings (1646 fuel assemblies).

When using the single heat exchanger SFPAHSF02, the system can maintain the spent fuel cooling water temperature at or below 51.3 °C when the heat exchanger is supplied with CCW water at the design flow and temperature. Using the heat exchanger SFPAHSF01 together with SFPAHSF03 would result in spent fuel cooling water temperature of 51.8 °C. The flow through the SFP provides sufficient mixing to maintain uniform water conditions.

If necessary to remove the complete core from the reactor (121 fuel assemblies) with the racks full from previous refuelings (1573 fuel assemblies) thus filling the racks to capacity (1694 fuel assemblies in total), the SFP cooling system is capable of maintaining the SFP water temperature at or below 72.4 °C when the SFP heat exchanger SFAHSF02 is supplied with CCW water at the design flow and temperature. The use of the heat exchanger SFAHSF01 together with SFAHSF 03 would result in a higher temperature of the spent fuel cooling water, i.e., 73.5 °C.

System piping is arranged so that failure of any pipeline cannot drain the SFP below the water level required for radiation shielding. A depth of 3.05 m of water over the top of the stored spent fuel assemblies is required to limit direct radiation to 2.5 mR/h (10CFR Part 20 limit for unrestricted access for plant personnel).

The SFP pump suction connections are located four feet below the normal water level and the cooling water return line contains an anti-siphon hole. These design features assure that the SFP cannot be drained more than four feet below the normal water level (normal water level is approximately 7.3 m above the top of the stored spent fuel).

A makeup water system is provided to replace evaporative and leakage losses. It consists of supplying demineralized water from the demineralized water storage tank upon a low-level signal from the SFP level instrumentation. Should the makeup water system not be operable, the secondary source of water would be from the reactor makeup water storage tank. The reactor makeup water storage tank is a Seismic Category I supply source.

Heat removal from the SFP can be achieved basically in two ways; i.e. through SFP heat exchanger or through evaporation of water from SFP or combination of both. In the case that the operation of heat exchanger cannot be achieved, the only way is through evaporation of water with boiling. In this situation boron remains in the SFP and there is no concern about criticality.

Enough amount of water needs to be provided to replace the evaporation.

Alternative means for establishing spent fuel pool makeup are:

- Pumping water from water pretreatment tanks with portable fire pump to the system for purification of SFP water surface.
- Providing water from fire protection hydrant network to the system for purification of SFP water surface. This method requires pressurized fire protection hydrant network by installed diesel fire pump.
- Pumping water from pool near water pretreatment building with portable fire pump to the system for purification of SFP water surface.
- Pumping water from circulating water intake pool with submersible fire pump and fire track to the system for purification of SFP water surface.
- Pumping directly to SFP with hoses from fire protection system.

If water in the SFP is decreasing even if makeup to the SFP is established, then operators are instructed to establish water spray over the spent fuel before the water drops below 3.05 m above the spent fuel elements. The priority of water sources is prescribed as follows: fire protection hydrant network, water pretreatment tanks, carbonate mud pool, circulating water intake and circulating water outlet pool.

Instrumentation is provided to measure the water temperature in the SFP, and to give local indication as well as annunciation at the main control board when normal temperatures are exceeded. Instrumentation is provided to measure water level in the SFP and give an alarm in the control room when the water level in the SFP reaches either the high or low level set points (15 cm above or 16 cm below the normal water level, i.e. 115.00).

SFP level measurement covers entire span from normal level to the bottom of SFP (i.e. 12.12 m). Temperature is measured at two different levels. Level and temperature indications are on local panel and on process information system.

Local panel with level and temperature indications is located in auxiliary building (AB) elevation 115 and accessible by stairs in AB building Thermo elements are RTD type and inserted in tube what assures accurate temperature measurement at selected points.

Level indication is ultrasonic type. This type of level indication can be inaccurate at high water temperature with steam at water interface and indication can oscillate, what is additional signal for boiling in SFP.

No cliff edge effects have been identified for period more than seven days because usage of alternative equipment assures spent fuel heat removal. All alternative equipment can be connected to the various systems through installed connection points by equipment operators from shift crew and firemen from fire team on site in less than one hour. Regular trainings and drills for shift crews and fire team are conducted periodically.

5.2.3.2 External actions foreseen to prevent fuel degradation

External support from an outside organization is not expected and not necessary. Equipment which is provided by the design can be used. Filling of the SFP is performed by the Krško NPP operating personnel who are normally on shift. All actions are performed according to the operating procedures.

5.2.3.3 Measures which can be envisaged to increase robustness of the installation

- Alternative cooling can also be established with installation of 8'' tee on the existing 24'' ESW line to CCW heat exchangers to provide alternative connection for fire protection pump with higher capacity and connection size of 8''. The capacity of already ordered pump »HFS HydroSub 450 floating unit«, is 720 m^3/h, which would provide enough heat removal of the decay heat from the SFP.
- New water line from Krško HPP which is under consideration would provide an alternative way of cooling the CCW heat exchanger. Gravity force would be used as a passive cooling system.
- Installation of fixed piping around the SFP with spray nozzles and connections for portable pumps will be considered.
- An alternative system with skid mounted pump and heat exchanger to cool the SFP.

5.2.4 Loss of the primary heat sink and loss of the ultimate heat sink

Krško NPP has primary heat sink which is also ultimate heat sink. Loss of the primary heat sink is described in section 5.2.3.

5.2.5 Loss of the primary heat sink, combined with station blackout

5.2.5.1 Design provisional autonomy of the site before severe accident

Design provisional autonomy of the site before severe accident is described in section 5.2.3.1.

Loss of the primary heat sink, combined with station blackout reduce the capability of using the existing as designed equipment, which needs to have electrical power supply and needs to be cooled.

Heat removal from the SFP can be achieved basically in two ways; i.e. through SFP heat exchanger or through evaporation of water from SFP or combination of both. In the case that the operation of heat exchanger cannot be achieved, the only way is through evaporation of water with boiling. In this situation boron remains in the SFP and there is no concern about criticality.

Enough amount of water needs to be provided to replace the evaporation.

Alternative means for establishing spent fuel pool makeup:

- Pumping water from water pretreatment tanks with portable fire pump to the system for purification of SFP water surface.
- Providing water from fire protection hydrant network to the system for purification of SFP water surface. This method requires pressurized fire protection hydrant network by installed diesel fire pump.
- Pumping water from pool near water pretreatment building with portable fire pump to the system for purification of SFP water surface.
- Pumping water from circulating water intake pool with submersible fire pump and fire track to the system for purification of SFP water surface.

- Pumping directly to SFP with hoses from fire protection system.

If water in the SFP is decreasing even if makeup to the SFP is established, then operators are instructed to establish water spray over the spent fuel before the water drops below 3.05 m above the spent fuel elements. The priority of water sources is prescribed as follows: water pretreatment tanks, fire protection hydrant network, carbonate mud pool, circulating water intake and circulating water outlet pool.

SFP level measurement covers entire span from normal level to the bottom of SFP (i.e. 12.12 m). Temperature is measured at two different levels. Level and temperature indications are on local panel and on process information system.

Local panel with level and temperature indications is located in AB elevation 115 and accessible by stairs in AB building Thermo elements are RTD type and inserted in tube what assures accurate temperature measurement at selected points.

Level indication is ultrasonic type. This type of level indication can be inaccurate at high water temperature with steam at water interface and indication can oscillate, what is additional signal for boiling in SFP.

SFP temperature and level instrumentation is normally powered from the process information system. In case of loss of normal power, the dedicated battery is provided to power SFP temperature and level instrumentation.

No cliff edge effects have been identified for period more than seven days because usage of alternative equipment assures spent fuel heat removal. All alternative equipment can be connected to the various systems through installed connection points by equipment operators from shift crew and firemen from fire team on site in less than one hour. Regular trainings and drills for shift crews and fire team are conducted periodically.

5.2.5.2 External actions foreseen to prevent fuel degradation

External support from an outside organization is not expected and not necessary in the early phase of the event (first 72 hours). Equipment stored onsite would be used. All necessary actions can be performed with shift crew and additional personnel from TSC and OSC. Organization for emergency situation would be established according to the RERP and EIP procedures.

For long-term operation, external support would be needed for diesel and gasoline supply to run the portable alternative equipment.

External support could be provided in offering enough potable water from the city of Krško or any other available water source.

All actions will be performed by shift crew onsite based on guidance in operating procedures

5.2.5.3 Measures which can be envisaged to increase robustness of the installation

- Alternative cooling can also be established with installation of 8" tee on the existing 24" ESW line to CCW heat exchangers to provide alternative connection for fire protection pump with higher capacity and connection size of 8". The capacity of already ordered pump »HFS HydroSub 450 floating unit«, is 720 m^3/h, which would provide enough heat removal of the decay heat from SFP.
- New water line from Krško HPP which is under consideration would provide an alternative way of cooling the CCW heat exchanger. Gravity force would be used as a passive cooling system.
- Installation of fixed piping around the SFP with spray nozzles and connections for portable pumps will be considered.
- An alternative system with skid mounted pump and heat exchanger to cool the SFP.

6 Severe accident management

6.1 Organization of the operator to manage accidents and possible disturbances

The emergency preparedness and response in Slovenia to eventual accidents at Krško NPP is conducted on plant, local, regional and state level. The Krško NPP is competent and responsible for on-site (site protected area) emergency preparedness and response including control over the exclusion area – an area from the reactor center out to a radius of 500 meters. Krško NPP maintains Radiological Emergency Response Plan (RERP). The Krško NPP’s RERP is coordinated with the RERPs of municipalities Krško and Brežice, the RERP of Posavje region and the Republic of Slovenia RERP. For the purpose of off-site emergency planning, and effective response, the following emergency planning zones (EPZ's) are specified around the nuclear power plant:

- precautionary action zone – is the area within 3 km around the plant where urgent protective action are planned and will be implemented immediately upon declaration of general emergency; the border of this area is determined by the borders of the local communities which lie in this area;
- urgent protective action planning zone - is the 10 km area around the plant where preparations are made to promptly implement urgent protective measures (sheltering, evacuation, stable iodine prophylaxis etc.); the border of this area is determined with the border of hamlets in Krško municipality and Brežice municipality;
- long-term protective action planning zone - is the 25 km area around the plant including the urgent protective action planning zone where preparation for effective implementation of protective actions to reduce the long-term dose from deposition and ingestion are developed; the border of this area is determined by the borders of municipalities which lie in this area; this area is also extended over the part of the territory of Croatia.

Evacuation times have been estimated for different areas, times and weathers conditions in which the population within urgent protective action emergency planning zone (10 km EPZ) surrounding the NPP Krško would evacuate in case of radiological emergency in NPP Krško. The evacuation time estimates have been developed accordance to the US nuclear power plants practice, and specific computer simulation model. The evacuation time estimates (ETE) includes the times for completion the following evacuation's actions:

- public notifications,
- preparation and mobilization and,
- actual movement out of the EPZ (i.e. on road travel time, including delays associated with vehicles queuing).

The ETE have been developed for 20 different combinations of local subarea (communities) inside the 10 km EPZ.

Krško NPP’s EPZs are shown in Figure 8.

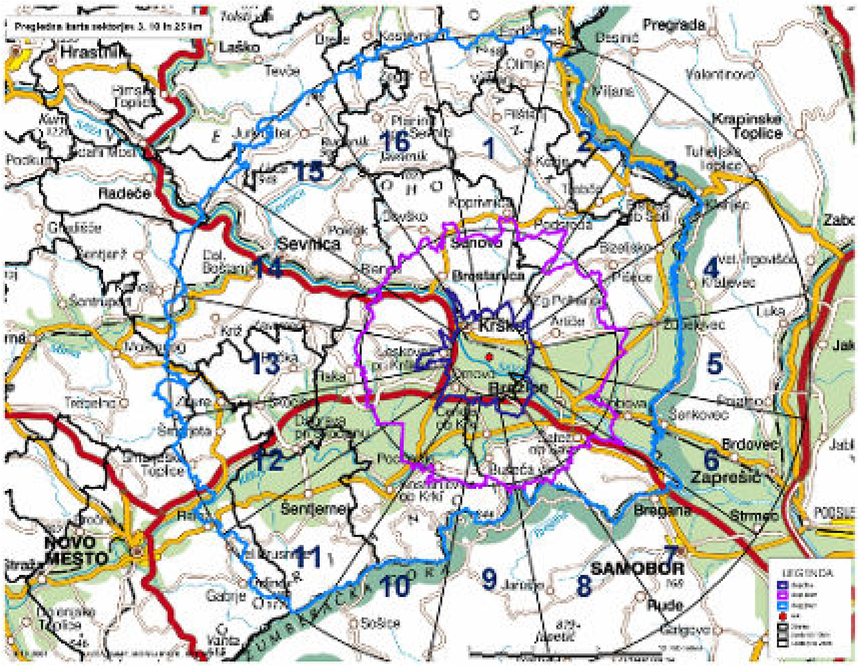


Figure 8: Krško NPP emergency planning zones

Krško NPP's emergency preparedness considers a wide range of postulated accidents from events where the radiological effect to the plant and the environment is negligible, to highly unlikely severe accidents, which could seriously affect the plant and environment. The radiological emergency classification methodology considers four levels of emergency from the lowest to the highest: unusual event, alert, site area emergency and general emergency.

- UNUSUAL EVENT – an event is in progress or has occurred which means a potential degradation of the plant safety or indicates a security threat to facility protection. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs.
- ALERT – events are in progress or have occurred which involve an actual or potential substantial degradation of the level of plant safety or a security event is in progress that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of a hostile act. Any releases are expected to be limited by small fractions of the protective actions intervention levels.
- SITE AREA EMERGENCY – events are in progress or have occurred which involve an actual or likely major failure of plant functions needed for protection of the public or security events that (1) result in intentional damage or malicious acts toward site personnel or equipment that could lead to likely failures or, (2) prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels exceeding protective action intervention levels beyond the site boundary.
- GENERAL EMERGENCY – events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with a potential for loss of containment integrity, or security events are in progress that could result in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed protective action intervention levels off-site for more than site area boundaries.

The classification is based on site specific emergency action levels (EAL) derived from generic examples of initiation events for each level of emergency and methodology used in US nuclear power plants. Emergency classification considers internal initiating events (fuel status, radiological effluent indications, primary leak to LOCA, primary to secondary leak, loss of power events, loss of indication events, safety systems anomaly, primary systems anomaly, secondary systems anomaly, other abnormal plant conditions, fire and personnel radiation injuries), external initiating events (earthquake, high winds or tornado, flood, low water, man made external events) and security contingencies.

The Krško NPP is responsible for the emergency classification, Emergency classification is important for timely declaration of the accident, initiation of on-site and off-site preventive protective measures (e.g. site evacuation in case of site emergency and preventive evacuation of 3 km EPZ in case of general emergency), activation of onsite and offsite emergency response organization (ERO) to the extent required and initiation of appropriate accident mitigation and emergency response measures.

6.1.1 Organization planned

Krško NPP maintains emergency preparedness as the constituent element of the entire nuclear safety concept and as the integrated part of the plant's internal organization structure and working process.

For maintaining the emergency preparedness, lines of responsibility are defined inside the plant's internal organizational structure. The president of the management board is responsible for the overall Krško NPP emergency preparedness. He approves Krško NPP's RERP and assigns the personnel to Krško NPP's emergency response organization (ERO). In case of an emergency he acts as an Emergency Operations Facility (EOF) director. The Technical director is responsible for keeping the plant in an overall safe condition. In case of an emergency he acts as an Emergency director. The Engineering services director is responsible for the overall Krško NPP's emergency preparedness planning. Overall emergency preparedness is conducted as plant's program under the Engineering Service Division. The particular emergency preparedness elements are conducted in different departments and through different plant programs (the Radiological environmental monitoring program in Radiation Protection Department, the Fire protection program in Operations Department, Training program in Training Department, Security program in Security Department etc.). These particular programs are in coordination with overall Emergency Preparedness Planning Program. Krško NPP's accident management and emergency planning respects national legislation and European directives as well as considers US practices and international guidance and industrial experience. The adequacy of Krško NPP's emergency preparedness is regularly evaluated and controlled through integrated exercises, independent internal audits, inspections, periodic safety reviews, international missions etc.

The entire concept of the onsite emergency preparedness and planning is determined in Krško NPP's RERP. It is a license document, reviewed at least annually and approved by the Krško NPP management board. The main objectives of the Krško NPP's RERP are:

- identification and evaluation of various types of accidents and emergencies which could potentially occur at the plant;
- identification of onsite accident management and emergency response measures;
- identification of onsite emergency response organization (ERO) and the responsibilities for the overall command, control and coordination of the emergency response;
- identification of responsibilities to carry out particular emergency measures;
- delineation of offsite support;
- delineation of obligations of Krško NPP regarding offsite emergency response;
- delineation of coordination of response activities with offsite authorities;
- identification of Krško NPP's emergency facilities, equipment and communications;
- identification of onsite recovery measures and organization;
- identification of basic instructions for maintaining onsite emergency preparedness.

6.1.1.1 Organization of the operator to manage an accident

The Krško NPP's accident management and emergency response organization (ERO) is an element of the site emergency preparedness. It ensures:

- overall direction and coordination of the onsite emergency response and co-ordination with offsite emergency response authorities;
- effective realization of particular emergency response measures.

Krško NPP's ERO is based on normal internal operating organization and consists of organizational structures activated depending on the emergency level and located in emergency response facilities (ERF). Organizational structure of the ERO (see Figure 9) is as follows:

a. main control room (MCR) and shift organization
b. onsite part of ERO (Technical support center (TSC) and Operational support center (OSC), which is activated in alert or higher level of an emergency;
c. offsite part of ERO (Emergency operations facility (EOF)), which is activated in case of site emergency or general emergency; EOF is located about 100 km from the plant in Ljubljana City where is also the location of Civil Protection Headquarters of the Republic of Slovenia (CPHRS).
d. additional support to the Krško NPP's ERO which is provided on contractual basis by offsite support organizations regardless of the emergency level.

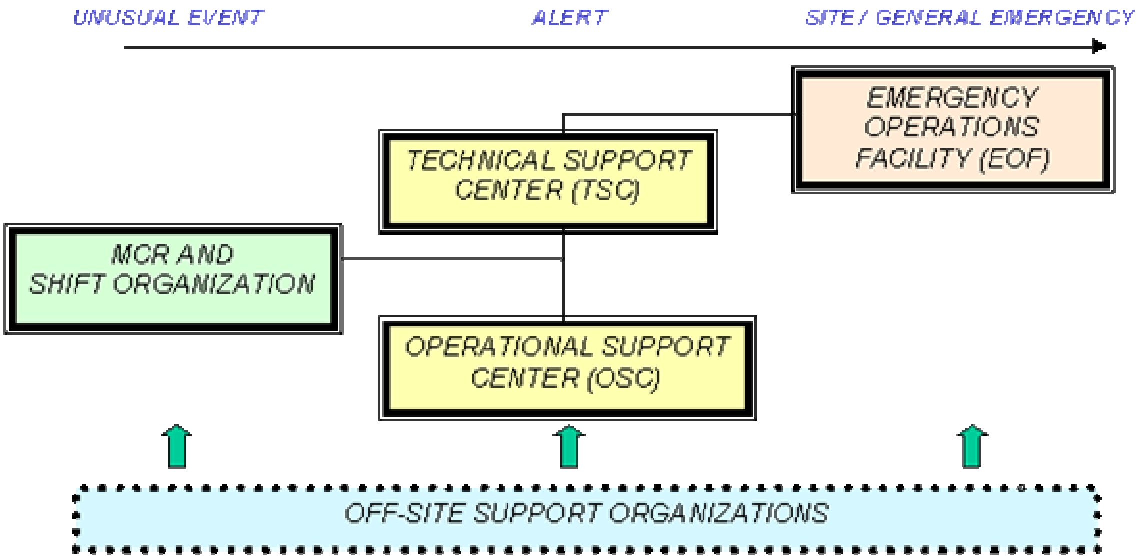


Figure 9: Krško NPP's emergency response organization

Krško NPP's ERO is established such that immediate emergency response and long-term emergency management are assured. Initial response is provided by predetermined emergency tasks and responsibilities of the normal shift organization. Further emergency response, continuity and intensity of emergency management are provided by activation of additional staff and the transfer as well as extension of tasks to the whole Krško NPP ERO. This assures immediate, efficient, linked and gradual emergency response appropriate to the emergency level.

Krško NPP's ERO comprises personnel possessing adequate technical knowledge and skills assuming their duties in accordance to the RERP. Emergency tasks and duties of intervention personnel correlate with normal organization tasks and duties. For each position in the ERO, emergency responsibilities and tasks are defined in the Krško NPP's RERP and Emergency Implementing Procedures (EIPs). In order to minimize confusion and assist in the control of the emergency response, the ERO is designed so that only dedicated person, or his alternative, is responsible for the implementation of specific emergency actions. In addition, the functional areas of responsibility remain flexible enough to accommodate the needs of the emergency and the availability of personnel.

The management board of Krško NPP assigns the personnel to the ERO in accordance with the selection criteria. The ERO is staffed with at least two persons at each position to ensure continuity of the response on activation, to have provisions for a dual shift operation which provide 24-hour coverage of emergency positions and enable long-term shift work. The Administrative Coordinator in the TSC and administrative coordinator in the EOF are responsible for assuring continuity of resources while emergency conditions exist. More than half of Krško NPP's workers (more than 300 individuals) are assigned to the Krško NPP ERO.

Krško NPP's ERO activation means alert and gathering of intervention personnel in Krško NPP's emergency centers. The shift personnel are available immediately after the emergency occurs. The response times of individuals assigned to the emergency organization during various weather and traffic conditions were determined as a result of actual Krško NPP's ERO activation drills. The response time of first essential intervention staff is 30 minutes after alerting. The TSC is able to carry out its emergency functions in about 1 hour after alerting. The EOF is able to carry out its emergency functions in about 2 hours after activation.

Krško NPP's ERO can be activated as follows:

1. Alerting individuals on mobile telephones using the software application installed on PC in MCR and in TSC. Dedicated commercial telephone lines are used for this application. The system enables alerting of a great number of individuals in a short time (approximately 100 persons in 30 minutes) with a possibility to receive a return notice of their availability. The shift engineer is responsible for activating the ERO by selecting the right scenario in RR in accordance to the emergency level declared. The whole Krško NPP ERO - all designated individuals (holding the same position) are activated at the same time. This assures wider initial response and availability of the intervention personnel. The long-term shift emergency management is coordinated after gathering personnel in emergency facilities.
2. If such ERO activation is not possible, the individuals are alerted by regular telephone calls in accordance with the activation list.
3. If telephone network does not work, self activation or alerting the intervention personnel over the media takes place.

The operability of software application for alerting emergency response organization personnel is checked every day. The response of ERO is checked monthly and is not announced in advance.

6.1.1.1.1 Staffing and shift management

Main control room (MCR) and Shift organization consists of 15 individuals as follows: shift supervisor, shift foreman, three reactor operators, four local equipment operators, shift engineer, three professional fire-fighters, radiation protection technician, chemistry technician.

In the case of unplanned shutdown or reactor trip, predetermined technical and engineering personnel are activated for further coordination of corrective actions and repairs.

In the case of an unusual event, the following positions are also activated: operations support coordinator (additional shift supervisor) to support MCR actions, public information coordinator to coordinate public information actions and management positions to relief shift supervisor of emergency director function, to attend the progression of the event and to support the shift crew in coordination of emergency response.

The emergency shift and unusual event organization is shown in Figure 10.

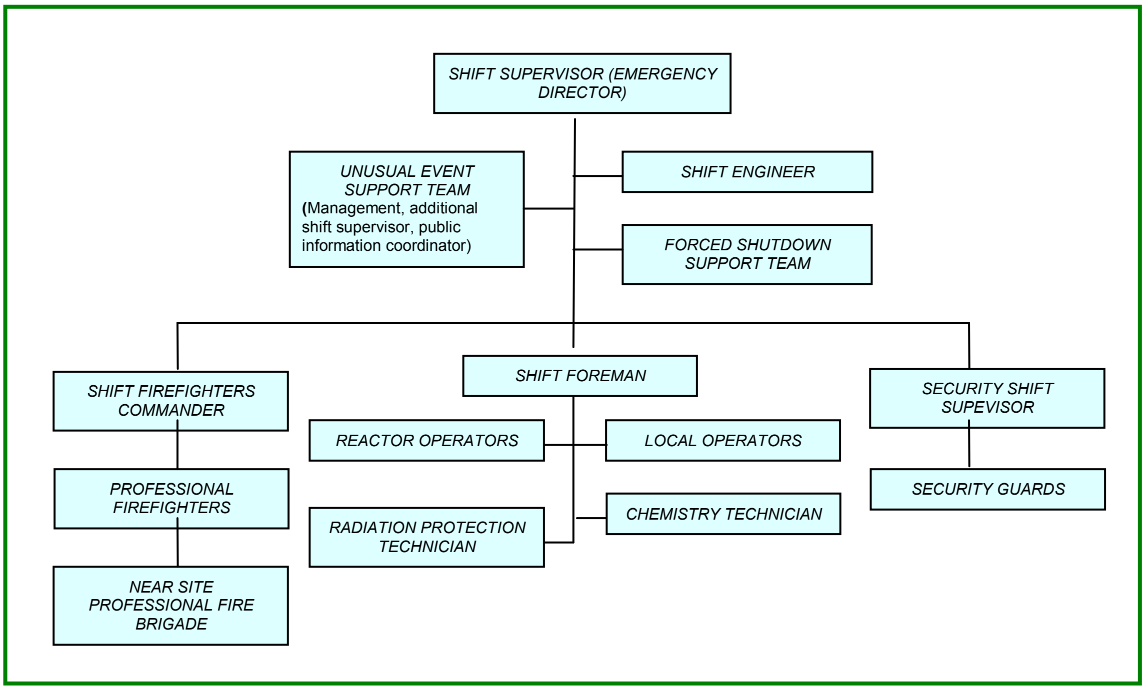


Figure 10: Krško NPP emergency shift and unusual event organization

Until the TSC is operable, management and coordination of the emergency response is organized within the MCR shift organization. The shift accident management and emergency response extend operators actions, corrective, protective and security measures, emergency classification, activation of ERO, off-site notifications, evaluation of off-site radiological consequences and protective actions recommendations. The Shift Supervisor assumes the function of the Emergency Director until either the Technical director or his alternate in the TSC assumes this function, or a close out of the emergency is declared. In case the shift supervisor becomes accidentally unavailable, the line of his alternative is determined. The shift engineer provides support to the shift supervisor in emergency response measures. In case the MCR has to be evacuated, the cool down of the plant is maintained from the evacuation panels. Shift supervisor and shift engineer evacuate to the TSC and direct the emergency response from TSC.

Technical support center (TSC) organization

The TSC is organized, equipped and structured to carry out the following emergency response functions:

- overall managing and coordination of on-site emergency response;
- evaluation of plant safety and emergency response status;
- operations and activities in MCR support;
- onsite technical and engineering support and coordination;
- making decisions and coordination of corrective actions;
- coordination of security measures;
- making decisions and coordination of protective and rescue measures;
- coordination with offsite support organizations and authorities;
- evaluation and making decisions on severe accident management strategies;
- coordination of logistics;
- provided EOF functions until EOF becomes operable.

The Emergency director directs and coordinates on-site emergency response. He is competent to make a final decision on the use of all available on-site resources for effective emergency response and to ask for additional off-site support. Non delegated emergency director responsibilities are: decisions on intervention personnel exceeding dose limits, decisions on severe accidents management strategies realization, decisions on plant evacuation and other important protective and corrective actions. The emergency director also assumes the functions of EOF director until this position is established. The line of emergency director's deputies is determined for cases when the primary individual on this position is not available.

The Emergency director is supported by:

- operations coordinator regarding coordination of activities with MCR, evaluation of operating conditions, preparing operating instructions, conducting fire fighting activities;
- onsite technical and engineering support coordinator regarding the plant and core status evaluation, TSC operation, technical and engineering evaluations and support, emergency classification, offsite notifications, evaluations of severe accident management strategies;
- radiation protection coordinator regarding control of radiation exposure and dosimetry, radiological survey, evaluation of offsite radiological consequences and coordination of offsite radiological monitoring;
- chemistry coordinator regarding radiochemistry and chemistry sampling and measurements, water treatment, decontamination and radwaste managing;
- maintenance coordinator regarding coordination of corrective measures and repairs on plant systems, structures and components and managing the OSC;
- public information coordinator regarding public information activities;
- security coordinator regarding evacuation, personnel accountability, plant access control, control over the site and exclusion area and coordination of other security activities;
- other TSC support personnel regarding implementation of individual emergency response functions.

The representative of the Slovenian Nuclear Safety Administration (SNSA) is present in the TSC.

The TSC organization is shown in a Figure 11.

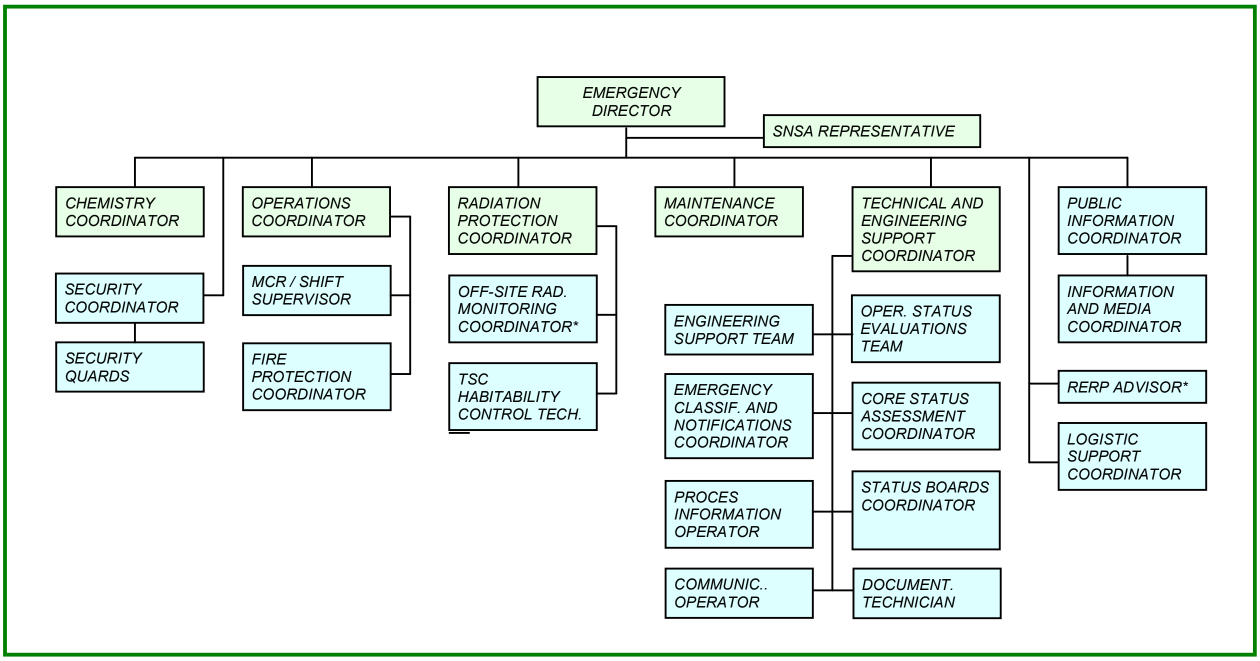


Figure 11: Krško NPP TSC organization

Operational support center (OSC) organization

The OSC is organized, equipped and structured to place the intervention teams and carry out the onsite intervention measures determined in the TSC. The following intervention teams are placed in OSC:

- radiation protection intervention team;
- chemistry intervention team;
- first aid intervention team;
- fire fighting intervention team;
- mechanical maintenance intervention team;
- electrical maintenance intervention team;
- I&C maintenance intervention team;
- other support personnel.

The OSC Coordinator coordinates OSC operation. He is responsible to the Maintenance coordinator in the TSC. Commanders of the intervention teams and other OSC support personnel provide support to the OSC coordinator. Commanders of intervention teams are responsible to coordinators of individual tasks in the TSC.

The OSC organization is shown in Figure 12.

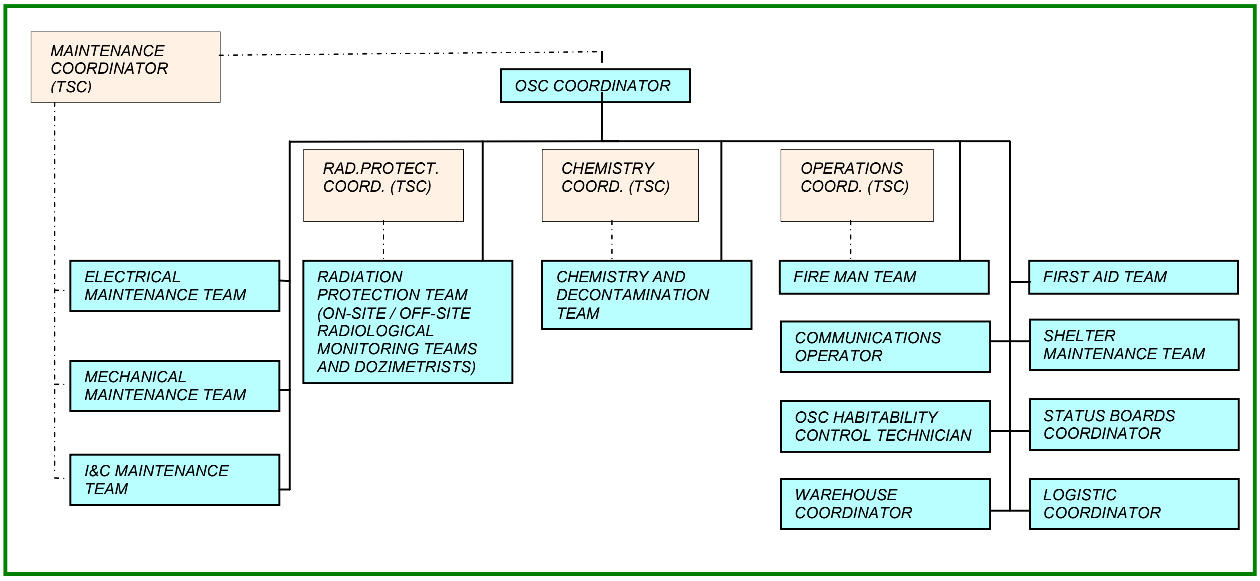


Figure 12: Krško NPP OSC organization

If TSC and/or OSC primary locations are not available, TSC/OSC is evacuated to the other onsite locations or to the near-site alternative location. The alternative location also provides the opportunity to gather TSC and OSC personnel if access to the plant is impeded. In this case the shift supervisor assumes the emergency director's function and TSC emergency response functions are delegated to the MCR and EOF. In such a case, the EOF is activated irrespective of emergency level declared. The alternative location is a predetermined reserve location about 3 km from the plant, equipped, sized and organized to place the TSC and OSC staff to be in readiness for intervention immediately after the access to the plant is established. The arrangements, decision making responsibilities and functional criteria regarding TSC and OSC evacuation are delineated in a special procedure.

Emergency operations facility (EOF) organization

The EOF is organized, equipped and located to carry out the following emergency response measures:

- overall direction and co-ordination of the Krško NPP's emergency response;
- engineering, technical, logistic and other support to the TSC and intervention personnel onsite;
- coordination with the Civil Protection Headquarters of the Republic of Slovenia (CPHRS), SNSA, other authorities and offsite support organizations;
- emergency classification and offsite notifications;
- evaluation of offsite radiological consequences and recommendations of urgent protective measures for the population;
- public information.

The President of Krško NPP's Management Board assumes the function of EOF director. The EOF director manages and coordinates the overall Krško NPP's emergency response. He coordinates the activities with the Civil Protection Commander of the Republic of Slovenia, SNSA director and other state authorities. The EOF director is competent to make a final decision on the use of all available Krško NPP's resources to manage on-site emergency. Non-delegated responsibilities of the EOF Director are emergency classification, offsite protective actions and offsite authority notifications. He is responsible for recovery measures after the emergency close-out. The line of EOF director's deputies is determined in case the primary individual on this position is not available.

The positions in EOF supporting EOF Director are:

- emergency director in TSC regarding direction and coordination of onsite emergency response;
- offsite dose assessment coordinator regarding evaluation of offsite radiological consequences, offsite urgent protective actions and coordination of these activities with offsite organizations;
- EOF engineering and technical support coordinator regarding plant status evaluation, engineering and technical support coordination, EOF operation, emergency classification, notifications and coordination with offsite authorities (SNSA);
- public information coordinator regarding public information activities, rumor control, media coordination and coordination of these activities with offsite organizations;
- logistic coordinator regarding logistic support coordination;
- other support personnel in EOF regarding implementation of emergency response functions.

If the TSC is not able to perform its emergency response functions, e.g. in case of the evacuation to the near-site alternative location, EOF has manpower and possibilities to take over some TSC's functions (such as plant status evaluation, severe accident management strategy evaluation and determination, operational support to plant operators etc.).

Representatives of SNSA are also present in EOF.

The EOF organization is shown in Figure 13.

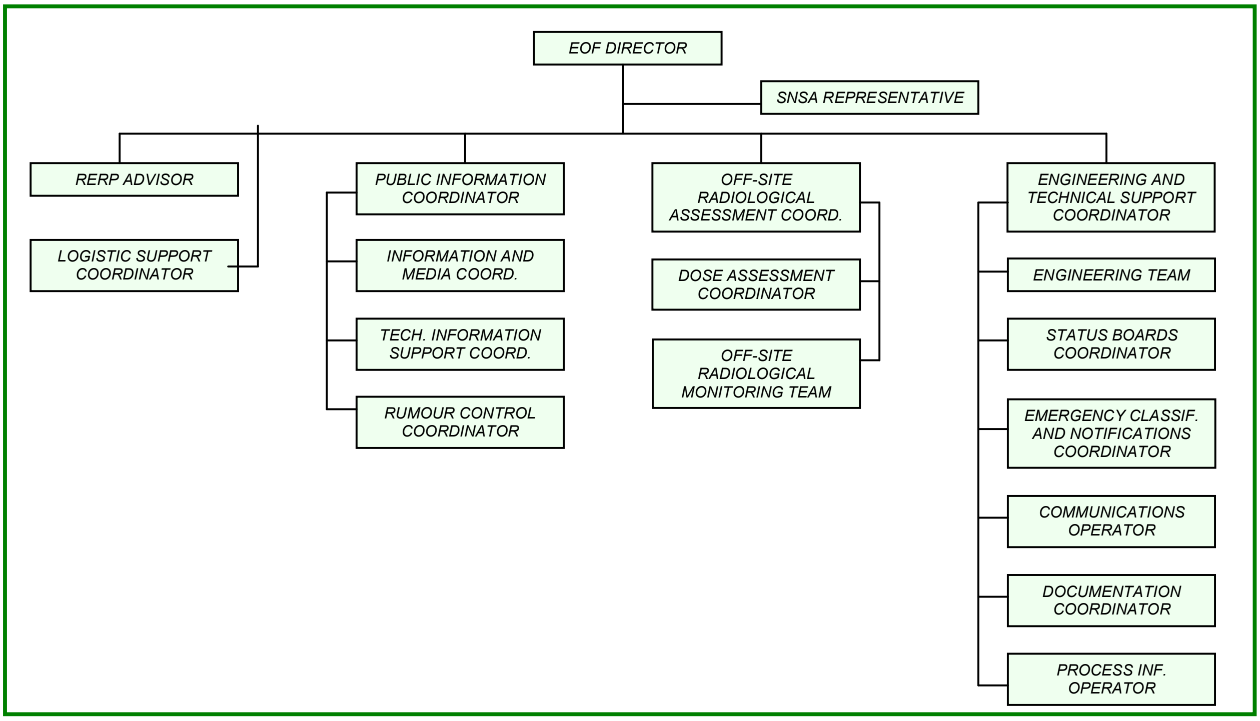


Figure 13: Krško NPP EOF organization

6.1.1.1.2 Measures taken to enable optimum intervention by personnel

Emergency response measures are determined in Krško NPP's RERP and specified in plant procedures. Competences and responsibilities are determined in the ERO for direction and coordination of the overall emergency response. Responsibilities and competences in the ERO are specified for decision making, initiation, coordination, preparation, control and implementation of individual emergency response measures. Responsibilities for essential emergency response decisions are delegated to ERO competent personnel and cannot be delegated to other positions (e.g. classification of emergency, offsite protective actions etc.).

The ERO intervention teams (including operators and security guards) are man-powered for shift turnovers during the interventions and for long-term emergency response. At all times, the Krško NPP has 6 shift crews of licensed operators, sufficient number of licensed shift engineers and other personnel with operations knowledge not directly working in the MCR. Intervention personnel is educated, trained and prepared for their emergency tasks. Emergency response responsibilities are delegated in the area of personnel's expertise.

Protection of intervention personnel during the emergency, the exposure and contamination control and dosimetry are considered as one of the most important parts of the emergency preparedness. The line of responsibility for decision making regarding the exposure control is specified. The Emergency Director is competent to approve exceeding normal operating dose limits when necessary to protect public and prevent event escalation.

During the accident, the intervention staff is located in the emergency response facilities (MCR, TSC and OSC), which are structured, equipped and organized to enable long-term habitability. Adequate protection of intervention teams is one of the check points in the procedure dealing with direction to interventions.

6.1.1.1.3 Use of offsite technical support for accident management

Offsite support and assistance to Krško NPP is provided by the local and other offsite support organizations. Contracts and letters of agreements have been developed to delineate outside company/agency assistance and services. The contracts and letters of agreement are reviewed annually to reaffirm assistance and to verify communications channels.

Support by offsite support organizations is assured on demand at any time regardless of the emergency level. Support to Krško NPP has high priority over other activities of support organizations.

Krško NPP maintains the following offsite support for accident management:

- Providing near-site (within 3 km radius around the plant) radiological monitoring and laboratories analysis in case of radiological emergency. Long-term contract with institutes is renewed annually.
- Observing, providing, analyzing and forecasting meteorological conditions. Long-term contract with national agency is maintained.
- Providing intervention medical assistance to victims of non-radiological or radiological injuries and medical center support in case of a serious accident at the NPP. Long-term contract with local health service center is renewed annually.
- Maintaining preparedness to accept radiologically contaminated or over-exposed patients from the NPP requiring special medical treatment. Long-term contract with the medical center is renewed annually.
- Providing primary location for NPP evacuees control point. Long-term contract with the community is renewed annually.
- Providing alternative location for NPP evacuees control point. Long-term written agreement with gas electrical power plant is maintained.
- Assuring NPP's competence in the exclusion area. Long-term written agreement with the owner of the area around the NPP (plant's exclusion area) is concluded.
- Providing location for NPP's Emergency operations facility (EOF). Written notice with the national authority is provided.
- Providing transmitter location for the NPP's offsite radiological monitoring radio communication channel. Contract with local radio club is renewed annually.
- Providing location for TSC and OSC near site alternative location. Long-term contract with the local institution is renewed annually.
- Providing onsite and access road to NPP fire fighting and other intervention assistance (protection, rescue) in different emergencies (fire, flood, spill of flammables or dangerous substances). Long-term contract with local professional fire brigade is renewed annually. The professional fire brigade provides off-site support in 5 to 10 minutes after alerting. However, in case the fire brigade is hindered due to eventual catastrophic events, Krško NPP has onsite fire fighting manpower, equipment and sources to manage the fire until support from other professional fire brigades is available.
- Providing emergency technical and engineering support on plant status evaluations, accident condition analysis and corrective measures determinations. Contractual agreement with NSSS supplier is maintained.
- Ensuring priority separate (isle) offsite power supply to NPP from the nearest Gas Power plant Brestanica in case of loss of other offsite power. Agreement and operating instructions with relevant companies in the Republic of Slovenia are maintained.

Support to the plant can be also provided by the Civil Protection Commander of the Republic of Slovenia and by competent authorities in compliance with their competences and the national RERPs. This type of support is not planned in advance. It could include additional heavy mobile equipment (i.e. diesel generators, pumps, air compressors, mobile water tanks, mobile cranes etc.), fuel supply, additional protective and rescue equipment, logistic support, arrangements for medical treatment, transportation and other logistics support. This includes also the support of military.

Assistance to the plant in security related matters is provided through the national security program and the security plan.

6.1.1.1.4 Procedures, training and exercises

Accident management and corrective measures, individual emergency response actions and the activities for maintaining emergency preparedness are dealt with in detail in different types of Krško NPP procedures. The procedures are prepared, developed, revised, approved, distributed and recorded in a prescriptive manner in accordance with the Krško NPP document control program. Adequate procedures as well as other necessary documentation are available to the intervention staff in electronic and controlled hard copies in emergency response facilities. The intervention staff is regularly trained on the use of procedures and informed about procedure revisions. Responsibilities in the ERO are clearly defined as regards the procedure use and emergency response actions in them.

The main sets of procedures dealing with accident and emergency response are:

- abnormal operating procedures (AOPs),
- emergency operating procedures (EOPs),
- severe accident management guidelines (SAMGs),
- fire response procedures (FRPs),
- radiation protection procedures,
- security plan procedures,
- RERP implementing procedures (EIPs).

AOPs and EOPs are used by operators in the MCR to carry out operations actions on plant components and systems in case of abnormal or emergency operational conditions of the plant corresponding to design basis accidents (DBA) and beyond design basis accidents (BDBA) not involving core damage. Operation crew in the MCR is competent to take operational actions by EOPs.

The plant status evaluation team in TSC evaluates overall operational and safety status of the plant during an accident and supports the MCR crew as regards particular operations measures. In case of a severe accident when the EOP's are no more effective in preventing core damage the transition from EOP's to SAMG's is performed. Shift supervisor in the MCR makes a decision on the transition from EOPs to SAMG's based on transition criteria. The overall objective of the SAMGs is to terminate the severe accident condition so that three primary goals associated with SAMG's are achieved:

- to return the core to a controlled stable status;
- to maintain or return the containment to a controlled stable status;
- to terminate any fission product releases from the plant.

The plant status evaluation team in the TSC evaluates SAMG's and recommends severe accident management strategies to the emergency director. The emergency director makes final decisions on the implementation of particular severe accident management strategies. The SAMG decision making support group in the TSC (SAMG DMSG) supports the emergency director in making decisions about implementation of SAMG strategies. The following positions in the TSC are assumed within the SAMG DMSG: operations coordinator, technical support and engineering coordinator, maintenance coordinator and radiation protection coordinator. The plant status evaluation team monitors the effectiveness and positive and negative impacts of the implemented strategies and suggests appropriate corrective measures to the emergency director.

The EIPs are a set of procedures that have been written to effectively and efficiently implement a response to an emergency situation or conditions in accordance with RERP. The EIPs consist of six general categories of procedures that address classification of the accident, general response guidance, protective actions recommendations, emergency response facilities activation, emergency support activities and group support.

Emergency response training is a part of the overall Krško NPP training program coordinated by the training department. Training program is based on systematic approach to training process. Training consists of classroom, on the job training, drills and exercise. Emergency response training, activities are planned on annual basis within overall Krško NPP annual training plan.

The following target groups are regular involved in emergency response training plan:

- all Krško NPP's employees and contractors' personnel,
- intervention personnel assigned to the Krško NPP's ERO and
- intervention personnel of offsite emergency response support organizations.

Krško NPP employees and contractors' personnel become familiar with the RERP and emergency protective measures within performing of regular general employee training program.

The intervention personnel assigned to the Krško NPP ERO and intervention personnel of off-site emergency response support organization receive additional emergency response training for their respective assignments. The intervention personnel include shift staff (main control room operators, local operators, firefighters, security guards) and staff in Krško NPP's emergency response facilities (TSC, OSC and EOF). Emergency response training for these personnel consists of initial and continuous with specialized (proficiency) emergency response training.

The initial emergency response training is conducted for the individuals upon their assignment to the ERO. It includes general review of RERP and emergency procedures, and specialized training for their respective emergency response assignments.

Continuous emergency response training is conducted for ERO personnel annually and contains a general review of the RERP with the emphasis on particular themes and specialized emergency response training. This includes specific themes regarding the functions and tasks individuals have in ERO. The specialized training of main control room operators, shift engineers, TSC's and EOF's plant status evaluators and severe accidents management strategies evaluators and decision makers include also the emergency response procedures, severe accidents managements guidelines and procedures' backgrounders. The EOP's and SAMG's backgrounds training is exchange every two years. The specialized training of shift personnel (local operators, fire-fighters) and OSC's maintenance intervention teams include also knowledge about on-site severe accidents management mobile equipment and manipulations with this equipment. This personnel performs hands on training on the mobile equipment on a periodic basis, during their continuous training programs. Drills to accomplish tasks from procedures or on behalf of the plant evaluation team during annual emergency preparedness exercise are opportunities for other ERO personnel to obtain training on mobile equipment.

Krško NPP regularly conducts different types of drills and exercises to verify the status of emergency preparedness of ERO and participating support organizations, allow the participants to be familiar with their duties and responsibilities, develop and maintain skills, verify the adequacy of methods described in the emergency response procedures, check the availability and operability of emergency supplies and equipment, and to identify and correct erroneous performance. Adequate personnel with responsibilities for different tasks participate in drills, for example operators, fire fighters, security guards, RP technicians participate in fire-fighting drills, all onsite personnel participate in onsite evacuation drills etc.

The Krško NPP is carrying out the drills with the frequency as follows:

- offsite notifications – once per year (communications are checked once per month);
- fire-fighting – once per month;
- first aid (one shift per year) and medical intervention – once per year;
- offsite radiological monitoring – three times per year;

- assessment of offsite radiological consequences and protective actions recommendation – two times per year;
- post-accident sampling – once per year;
- post accident radiation monitoring – once per year;
- evacuation and personnel accountability – two times per year;
- activation of the ERO – once per year (the response of the intervention personnel is checked once per month);
- manipulation with onsite severe accident management mobile equipment and preparation for severe accident management strategies based on EOPs or SAMGs evaluations (on yearly basis).

The licensed operators are regularly trained in accordance to the licensed operator training program. It consists of four segments of training per year and includes operational management of plant abnormal and emergency conditions according to AOP's and EOP's on the plant's full-scope simulator. The scenario regularly includes accidents with the use of respiratory equipment in simulator control room and evacuation of simulator control room.

The elements of drills are included in integrated emergency response exercises. An integrated emergency response exercise is carried out annually to evaluate overall emergency response readiness of Krško NPP and participating organizations. The scenario of exercise is varied from year to year so that all major emergency response elements of the RERP are included in the exercise objectives and tested within a 5-year period. An exercise is carried out based on the scenario which in its final phase results in general emergency level, severe accident conditions and release of radioactive material to the environment, so that emergency response is needed in the plant vicinity as well. In a 5-year period the integrated national exercise is carried out with participation of local, regional and state emergency responders.

The full-scope real time Krško NPP's simulator serves as an exercise's scenario simulation tool for accidents (including severe accidents). The simulator is also used for the real MCR simulation.

6.1.1.2 Possibility to use existing equipment

All including alternative equipment, which is listed in the EOPs and SAMGs is located onsite. This includes fire fighting equipment, Health Physics and contamination control equipment, protective, rescue and first aid equipment, respiratory equipment, maintenance tools and instrumentation and other equipment for managing emergency under different severe conditions for longer period of time without offsite support.

Fire fighting equipment is placed in a fire fighting building. It includes fire fighting equipment for initial fire response, fire protective clothing, respiratory protective equipment and rescue equipment. The equipment is specified in fire protection program procedures and is regularly tested and adequately maintained. The equipment is placed on the location together with the shift fire-fighting team. The additional fire fighting equipment and support is provided from the professional fire brigade located about 2 km from the plant.

Health physics equipment is determined in USAR and is specified in radiation protection procedures. This equipment is stored in different locations mostly in facilities related to health physics, located in four areas inside the plant technological complex. Some health physics equipment (for example whole body counter) is also located in radiation protection laboratory, in administrative building. The health physics equipment includes protective clothing, respiratory protection equipment, air sampling equipment, decontamination equipment, fixed and portable radiation detection instruments and personal dosimetry devices. Sufficient quantities of each type of instrument permit calibration, maintenance and repair without diminishing the radiation protection supplied. The most direct reading dosimeters and TLDs are placed at the main radiological control point and in the radiological laboratory. Some are also available in emergency response facilities (MCR, TSC, and OSC) at fire fighters' location and at security guards location.

Portable shielding in the form of lead bricks and lead blankets are available in the plant. Most health physics equipment is operable under severe accidental conditions. In case of an emergency, the mobile radiological laboratory is dispatched to the surroundings of the plant for offsite radiological monitoring tasks. In case of severe radiological conditions, essential radiological instrumentation can be transferred from onsite to an offsite clean location.

The types and quantities of onsite respiratory protection equipment is listed in EIP's, fire protection and radiation protection procedures. Typical respiratory protection equipment includes:

- air purifying devices such as half- and full-face masks with combined filter cartridges;
- air supplying devices such as air line supplying devices (plastic suits with constant air flow) and self-contained breathing apparatus (SCBA).

The respiratory protection equipment is located in the fire fighters' building, health physics facilities, at security guards locations, in the MCR and in emergency response facilities (TSC and OSC). Krško NPP has enough SCBA for initial emergency response also in case of wider needs. Additional respiratory protection equipment (SCBA) is provided from the near site professional fire brigade.

Various types of protective clothing are stocked at the plant to protect personnel against contamination. Typical protective clothing includes protective clothing for body, head, hand and foot protection. The types and necessary quantities of protective clothing are listed in radiation protection procedures.

First aid equipment is located everywhere on-site. Additional equipment is available in the on-site health center, in the OSC and at two main locations inside technological complex. The first aid equipment also includes defibrillators. The first aid equipment is specified in EIP's.

The maintenance tools and instrumentation are placed in the central workshop in the administrative building, and in the hot workshop in the radiological control area. Spare parts are stored in two onsite warehouses and in the warehouse located 500 m from the plant.

The equipment is adequately dispersed on different plant locations. This assures partial availability in case one location is un-accessible because of accident conditions. The equipment is evident in plant procedures or plant electronic information systems which are still operable within a limited time after a loss of power supply. The equipment is adequately stored and maintained. The intervention staff knows the equipment location and is regularly trained on using it. The access to the equipment is also possible in accident conditions or in case of loss of power.

6.1.1.3 Provisions to use mobile devices (availability of such devices, time to bring them onsite and put them in operation)

It is estimated that the Krško NPP has man forces, mobile equipment and resources to manage initial emergency response in case of a severe accident for an extended time - up to 24 hours without any offsite support and up to 1 week with no needs for additional heavy mobile equipment from offsite. The mobile equipment essential for managing severe accidents (SAME) according to EOP and SAMG strategies are stored at different locations onsite. The SAME is placed on safe locations with respect to preventing their impairment in accident conditions (earthquake, floods, fire etc.). Mechanical connections, power supplies, connection tools and other arrangements are prepared in advance at locations and on components of systems where SAME should be connected to or applied to implement the required severe accident management strategies. This enables preparation and implementation of severe accident management strategies only with shift crews effectively trained for accident conditions.

The SAME (list of available SAME mobile equipment is in the appendix of the report) is included in Krško NPP equipment data base as an AE (Accident Equipment) system and is regularly tested and maintained in accordance to plant maintenance procedures. Regular training and drills for shift personnel and other personnel in ERO responsible for implementation of severe accident strategies and handling with the SAME are conducted on an annual basis.

6.1.1.4 Provisions for and management of supplies (fuel for diesel generators, water, etc.)

Diesel fuel storage capacity for standby emergency diesel generator units is supplied on the site to provide post-accident power requirements for seven days. Each standby diesel engine incorporates a separate 2.07 m^3 fuel oil day tank in its associated fuel oil transfer system. It is sufficient to fuel each diesel engine for a period of four hours, continuously running at full load. Additionally, the plant maintains onsite fuel oil supply for mobile diesel generators for a severe accident management operation of a period of 3 days. It is estimated that the plant could permanently maintain water supply – a loss of ultimate heat sink severe accidents could be managed for a period of 72 hours without additional offsite water supplies. A 5-ton boric acid supply onsite is maintained permanently. It is estimated that the stock of 3000 KI (potassium iodide) tablets is enough to protect onsite intervention personnel for a period of 2 weeks. Krško NPP maintains stores of other supplies (food, drinking water etc.) to function without offsite supplies for a period of days.

In accident conditions, additional supplies could be obtained through the national civil protection support. This support is not planned in advance.

6.1.1.5 Management of radioactive releases, provisions to limit them

Introduction

In case of an accident leading to nuclear fuel overheating and the possibility of release of radioactive material from the reactor to the containment, the main emergency safety features to limit the releases to the environment are the containment and its systems. Management of radioactive releases is performed by the containment systems and plant ventilation system with high efficiency particulate air (HEPA) filters and impregnated charcoal filters for iodine retention. Information important for this management are provided by instrumentation and control signals including radiation and effluent monitoring system, environmental radiation monitoring and meteorological parameters monitoring. These pieces of information together with reactor core status are used by release modeling and dose projection tool for realistic dose projection related to meteorological conditions and release source term. The results of calculated scenarios can be provided as an additional aid to emergency management and decision making.

Containment systems

Krško NPP has a large dry containment. The containment systems consist of the steel shell containment, concrete shield building, penetrations, and the directly associated systems upon which the containment functions depend, as follows:

a. The containment isolation system isolates various fluid systems that pass through the containment wall to prevent the direct release of radioactivity to the environment in the event of a postulated accident.
b. The containment spray system has a dual function:
c. Heat removal by spraying of borated water through the containment; water collected in the containment sump is returned to the containment spray system at the discretion of the operator.
d. To enhance fission product removal efficiency and prevent significant re-evolution of the dissolved iodine species back into the containment atmosphere as volatile iodine after the recirculation phase begins.
e. The containment air recirculation and cooling system maintains the containment atmosphere at or below the design pressure and temperature by transferring the containment heat to the component cooling water system. This serves to reduce the leakage of airborne radioactivity from the containment building following an accident.

f. A combustible gas control system is provided for the post-accident control of hydrogen in the containment. This is accomplished by processing the containment air through the electric hydrogen recombiners;

Design of major containment components

a. A leak tight steel containment vessel is designed to withstand the temperatures and pressures associated with postulated loss-of-coolant accidents and main steam line breaks as well as collapse pressures induced by inadvertent operation of the interior spray system.
b. An annulus space between the steel vessel and concrete building which is maintained at pressure less than atmospheric. This effectively prevents leakage of contaminated air to the outside. All air in this space is filtered.
c. The reinforced concrete shield building which provides the required biological shielding between the containment and outside spaces.
d. Double barrier penetrations at all pipe entrances which ensure prevention of air leakage around pipes while maintaining independent behavior between the shield building and the containment vessel.
e. Annulus negative pressure control system is designed to limit the maximum pressure in the annulus immediately after a LOCA and achieve a negative pressure differential in the annulus relative to the outside to minimize ground level release of airborne radioactivity due to containment vessel exfiltration during post-accident conditions.
f. Annulus filter system is designed to minimize offsite radiation exposure following design basis accident (LOCA). During LOCA and post-accident conditions both trains of this system start automatically and continue to operate to maintain the negative annulus space differential pressure. The main components include:
g. Two 100% capacity separated filter plenums, each including demister, roughing filter bank, electric heating coil, HEPA filter bank, charcoal filters and a second HEPA filter bank. The plenums are shielded for radiation protection. They are located in the auxiliary building. Fire and smoke safeguards have been provided through instrumentation and monitoring devices. Two 100% capacity exhaust fans are located in the auxiliary building.
h. Inside the containment, there is charcoal cleanup system with HEPA and charcoal filters designed only for normal operating conditions for cleaning the containment air before containment entry. It might be used also in case of post-accident operations.
i. Two redundant electric hydrogen recombiners are provided to sustain all normal loads as well as accident loads including seismic loads and pressure transients. Process capacity is such that the containment hydrogen concentrations will not exceed four volume percent following design basis event. The recombiner is manually controlled from a panel located outside the containment.
j. In addition to the recombiners, there is also installed hydrogen control system as a backup system. It is designed to retain its integrity and operability under all emergency conditions with the capability of purging the containment for long term hydrogen control following a design basis loss of coolant accident. This system operation and flow rate are manually controlled from the main control room and fan operation and isolation valve positions for both the exhaust and negative pressure relief trains are monitored from the control room. The exhaust train is electrically interlocked so that the exhaust fans cannot run unless the spent fuel pool charcoal exhaust system is operating and further interlocked so that isolation valves close on a containment isolation signal or on excess exhaust air flow. There is also available a redundant sampling system monitoring the containment atmosphere.

Accidental release monitoring

Radioactive releases are monitored during normal operation by standard radiation monitoring system (RMS) channels for noble gases, particulates and iodine. The samples are taken periodically and analyzed. Noble gas effluent sample from plant ventilation duct is also continuously analyzed for its composition. Noble gas monitors and effluent sampling became redundant after upgrading by post-accident radiation monitoring system (PARMS).

The PARMS consists of lower range warning monitors, accident range monitors and post-accident effluent sampling skid:

- Plant vent header monitoring and isokinetic sampling for normal and accident conditions;
- Secondary side condenser air ejector monitoring and sampling;
- Steam generator leakage monitoring;
- Containment high-range radiation monitoring;
- Failed fuel monitor at reactor coolant let-down pipe;
- Steam generator relief valve monitors;
- Auxiliary building ventilation exhaust monitor and sampling;
- Fuel handling building ventilation exhaust monitor and sampling;
- Fuel handling building high range area monitor.

Parallel to normal effluent sampling and monitoring unit there is a post-accident sampling unit with high range noble gas detector. The sample is provided by derivation tube from the main sampling pipe. Post-accident sampling unit is normally in stand-by. In case of high radioactivity of the gases, a solenoid valve opens a by-pass line to the main pump and the low range measuring branch is isolated by a motor controlled valve. The sample flow rate through particulate and iodine sampling skid is 1 l/min.

A filter cartridge (fibre glass and silver zeolith) in the sampling skid can be removed at the end of a preselected sampling time. Post-accident sampling unit is remotely controlled from the MCR. The filter cartridges and a filter trolley, used for transportation for subsequent analysis, are shielded by lead to protect the personnel. The system is located in auxiliary building. The components of the system and the sampling lines are installed and located in a way that the post-accident doses to the personnel operating post-accident sampling would be below 50 mSv.

Radioactivity inside containment in case of a severe accident can be assessed by high-range radiation monitors and by post-accident sampling system (PASS) of gases in the containment atmosphere.

Release modeling and dose projection tools

1. Procedural approach

Assessment of source term for possible release is based on plant data on core temperature, containment dose rate monitoring and other radiological sampling data if available. Individual release scenarios based on PSA studies of typical accidents and standard NUREG source terms have also been established. This specific assessment tool for radiological consequences in the environment has been prepared by the plant, partly upon request from regulatory body. It utilizes a more realistic dispersion model, Lagrangean instead of the simple Gaussian, to calculate dispersion in the environment. This is of importance for the areas with complicated meteorological modeling environment, such as Krško.

Graphical and numerical presentation of projected data is presented in several ways, for example: dose at different distances and exposure types and map containing resulting ground dose rates due to fall-out. Dispersion projection may also be viewed in a three dimensional model. Meteorological input data is automatically taken from the NPP environmental information system while input data on release source term is manually given based on emergency procedures as well as by automatic transmission through the plant process information system. The Programe can be also driven by manual data input and the regulatory body emergency assessment group has the same software.

2. Radioactivity assessment

Radioactivity of the nuclides in the reactor core is calculated in real time using measured reactor power data. About forty nuclides were selected to be important in potential in-containment release source term. Basic data for core damage assessment are provided from reactor core exit thermocouples, containment radiation monitoring, indication of reactor vessel level, and some verification to define extent of damage. These data are used in the plant status analysis according to the radiological emergency plan. There is also manual option for radioactivity assessment based on the design data or regulatory guidelines.

3. Capability of the effluent modeling tools

The central unit of the system is personal computer with software application providing the following:

- using of meteorological data from local environmental stations,
- display of environmental radiation monitoring data,
- using real time reactor power data for source term calculation,
- receiving information from plant effluent monitoring channels,
- manual options also for most automatic data inputs,
- core damage assessment based on basic information,
- definition of release source term based on measurements or safety assessments,
- dose calculation and presentation for early exposure pathways.

Meteorological data are provided from site meteorological tower and sodar unit. Additional or backup data can be used from three other meteorological stations around the plant or from state prognostic service. There is also possibility for these data input by a manual procedure. The system is installed on 25 km × 25 km domain in the orographic and land use resolution of 250 m. The data acquisition and modeling calculations are made every half an hour and are completely automated. Radiation monitoring data are refreshed every minute.

Offsite radiation monitoring

NPP Krško provides offsite radiation monitoring team to perform initial near site radiological measurements and sampling. Team provides data to the TSC or EOF to support the utility effort in radiological release assessment and off-site protective measures recommendations to the authorities. Deployment time for the offsite radiation monitoring team ranges from 45 minutes to 1 hour and 30 minutes from initial notification. Team is located in OSC and is dispatched to the surrounding of the plant in case of site or general emergency level declaration. It uses the mobile laboratory equipped with the instruments, sampling kits, analyzing equipment and communication means. Team has a capability of estimating iodine concentrations less than 3.7E3 Bq/m^3 range using portable equipment. The team works under the direction of the Radiation Protection Coordinator from TSC, or under the Offsite Radiological Consequences Assessment coordinator from EOF.

The main environment radiation monitoring in case of accident is coordinated by SNSA and is provided by professional national radiation monitoring teams, and by local civil protection organization's teams. Additional support could be provided also by national army's teams. The environment radiation monitoring is the part of the overall routine offsite environment survey program. The permanent routine offsite environment survey program is conducted by national institutions and consists of surveillance of the area extending to a radius of 12 km around NPP Krško, with the exception of the Sava river, where it is extended to 30 km downstream of the site. The following media are regularly surveyed: water, suspended matter, sediments and biota of the Sava river at two reference and two monitoring points; underground water in the wells supplying drinking water for cities of Krško and Brežice, as well as in some test boreholes located close to the Sava river in the vicinity of city of Zagreb. 8 aerosol and 7 iodine pumps placed up to a distance of 2 km around NPP Krško are used for continuous air sampling; wet and dry deposition is collected by 3 rainfall collectors and several »vaseline plates«; locally produced foodstuffs, folder, and flooded soil are analyzed seasonally. In case of accident with release of radioactivity or potential release, the routine environment survey program transfers to the emergency environment survey program with additional sampling and analysis, which are performed when either evaluating a situation to determine if emergency exists or during emergency response activities.

Level of radiation around NPP is also permanently monitored with 13 continuous dose rate monitors as a part of the overall national radiological early alerting system. 57 passive thermo luminescent dosimeters are also placed around NPP Krško.

<u>Offsite protective actions recommendations</u>

During the accident NPP Krško has to provide protective actions recommendations (sheltering, evacuation, use of KI tablets) for population in 10 km EPZ to the Civil Protection Headquarters of the Republic of Slovenia (CPHRS) commander. The protective actions recommendations have to be provided in 15 minutes after declaration of the emergency level. In case of general emergency level preventive evacuation of 3 km radius around NPP has to be recommended as initial offsite protective measure. Based on the plant condition, containment status and radioactivity levels, core temperature status and potential of the radioactivity release to the environment initial wider area preventive evacuation (up to 10 km radius around NPP) could be recommended. The initial assessment of potential for radiological release and offsite protective actions recommendations are done by shift supervisor in MCR. After full implementation of NPP Krško ERO environment radiological survey, results of filed monitoring and samples evaluation, offsite protective actions recommendations are provided by TSC Radiation Protection Coordinator, and by EOF Radiological Consequences Assessment Coordinator respectively. During the accident, core status is evaluated continuously by Core Damage Assessment Coordinator in TSC.

NPP Krško coordinates offsite protective actions recommendations with SNSA. Final decision on public and environment protective measures is taken by CPHRS commander. Regional and local authorities are responsible for implementation of public and environment protective measures.

6.1.1.6 Communication and information systems (internal and external)

See section 6.1.2.1 below.

6.1.2 Possible disruption with regard to the measures envisaged to manage accidents and associated management

6.1.2.1 Extensive destruction of infrastructure around the installation including the communication facilities

The main railroad Ljubljana - Zidani most - Zagreb - Belgrade with heavy traffic in both directions is located about 800 m north of the plant boundary. Along the railroad there is local road Krško - Brežice with moderate traffic. Highway Ljubljana - Zagreb with heavy traffic passes about 4 km south of the site and local road Krško - Drnovo - Brežice 3 km from the site. Access to the site is provided from the local road Krško to Brežice passing north of the site. An access railroad about

2 km long is constructed from Krško station to the site and connected to the Ljubljana - Zidani Most - Zagreb railroad network.

In the case of the external events described in the sections 2 and 3 of this document it can be expected that the normal access path to the site could be restricted. The distance from the nearest city Krško on the Northwest to site is less than 2 km and the distance to the city Brežice on the east is approximately 13 km. Most plant workers also live in the vicinity of the plant (not more than 10 km away from the site) consequentially it is estimated that sufficient number of emergency personnel could arrive on site in any credible circumstances.

In the case of extensive external event some aggravating circumstances could be expected regarding the plant emergency staff arrival to the site. It was estimated that the bridges over the Sava river present probably the weakest points regarding the access to the facility in the case of strong earthquake. However, there are many possibilities to cross the river from the different directions. One of the options is also to use the river dam structure at the site.

Protection against flooding of the plant was accomplished by the construction of the left side Sava dike and the fact that the excessive river water flow will be spilled over the right side river bed (which is 1 m lower) preventing the over flooding of the left side where the plant is situated (see Chapter 3 of this report). It is not expected that the probable flood will significantly limit free access to the plant from any direction.

The plant communication system provides facilities for several different kinds of information transmission between plant buildings and off-site locations. The following communication subsystems for internal plant communication are installed:

- Telephone system (dial telephone system) between all important locations on the plant (MCR, TSC, OSC, etc.),
- Wireless VHF radio system,
- Plant paging system,
- Sound powered telephone system.

On the other hand there are also different types of communication between the plant, Emergency Operations Facility (EOF) and other external organizations involved in the Emergency Preparedness Plan:

- Radio system (portable radio stations),
- Direct phone to the SNSA (independent of commercial dialing telephone system),
- Direct communication system to power distribution center,
- Mobile telephone system,
- Satellite phone between the plant, EOF and plant security department,
- Wireless VHF radio system to the local police station
- National protection and rescue’s radio system,
- National firefighting organization’s radio system.

All of these systems are powered from the different uninterruptible power supplies (UPS), however there are some provisions planned in the emergency operation procedures to secure alternative power supplies to the particular communication system in the case of prolonged Loss of all AC power (station blackout) onsite. The wireless and plant paging system are designed and purchased for the operation in rough industrial environment, however in the case of extensive earthquake (building damage) we can expected that some portions of the system would not be operable. Different and variable ways of communication between onsite buildings and offsite facilities (which are listed above) assure that there shall be no major loss during the postulated events. The main plant page system could also be powered from the alternative diesel generator from the switchyard.

The communication infrastructure of the plant also offers two different internet possibilities for communication with the local and other communities and organizations. Connection to the internet is provided via two physically separated fixed (optical) connections provided by two different providers. The first connection is routed via the high voltage transmitting lines and the other is utilized on underground optical connection. The second transmission line is also hosting independent high speed Ethernet connection (based on optical fibre) to the closed and secured SNSA network (MKSID) which is used for internal communication during emergency drills and events. Communication between NPP Krško's EOF which is located some 100 km away in the capital city Ljubljana and the plant are based on the dedicated telephone lines and high speed optical connection of the internal Process computer network (PCN) which is utilized on the first previously optical communication route via high voltage transmission lines. PCN is used for the remote usage of plant Process information system (PIS) in the EOF. The usage of PIS in the EOF enables monitoring of the plant parameters in the same way as the site. The usage of plant PIS is enabled of course also in both site emergency centers (TSC and OSC).

The PIS system onsite is supplied from non-safety 220 V DC batteries via three different UPS invertors sufficiently rated to supply the system for the first 4 hours after the initiating SBO. One of the priorities of the operation after these hours will be to enable supply of the PIS UPS's via the alternative (if the emergency diesel generators will not be available) diesel generators. The PIS system is not an essential system and it is used for monitoring purposes, but can significantly help the emergency management organization follow the progress of the accident to support MCR staff.

The plant operating staff excluding security (there are 15 technical individuals available all the time onsite) is able to implement the plant EOPs and other required actions from the site RERP by itself without additional support for at least 24 hours. All equipment listed to be used during a serious event is present onsite with the following supplies:

– Emergency diesel generators	Fuel oil supply for 7 days of operation
– Additional portable emergency equipment	Fuel supply for 72 hours of operation

However, in accordance with the RERP the time to activate and to achieve the operability of the plant emergency support centers (TSC and OSC) is 1 hour and 2 hours for the EOF. The number of the OSC staff is sufficient to support the implementation of all the needed corrective maintenance. On the other hand, operating staff is trained and capable of operating all prescribed emergency equipment by themselves or by the support of the fire fighters permanently present onsite (3 persons).

6.1.2.2 Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities on site

6.1.2.2.1 Impact on the accessibility and habitability of the main and secondary control rooms, measures to be taken to avoid or manage this situation

The habitability systems for the control room are designed with the following considerations so that habitability can be maintained under normal and accident conditions:

- To maintain the required ambient air temperatures in all control room areas for the comfort and safety of personnel and to meet environmental requirements of equipment.
- To adequately meet the requirements of 10CFR Part 50, Appendix A, General Design Criterion 19, to permit access and occupancy of the MCR under the postulated accident conditions.
- To provide a Control Room Emergency Charcoal Cleanup System for operation under abnormal conditions.
- To detect abnormal conditions such as high radiation which require the control room isolation from outside environment, which means the Control Room Charcoal Cleanup System actuation.

- To provide the control room with a smoke venting system capable of purging the control room with fresh outside air upon the detection of smoke.

The heating, ventilating and air conditioning (HVAC) and emergency filtration systems for the main control room consist of the following subsystems:

- MCR air conditioning system
- MCR charcoal clean-up system
- Chilled water generating and distributing system

During post-accident conditions, pneumatically operated dampers automatically isolate control building rooms from the outside atmosphere. The MCR air conditioning and electrical room cooling systems assume full recirculation. The MCR cleanup system is manually started, if required, during control room isolation to keep the area habitable.

The MCR isolation is initiated by either a safety injection signal or by a high radiation signal as detected by the radiation monitor in the MCR, or by isolation signal generated upon Hi-Hi Chlorine level detected on MCR-HVAC outside an air inlet structure. Isolation may also be initiated manually from a local control station.

The operator can add fresh air to the control room under post-accident conditions through the emergency charcoal cleanup system outdoor air intake by using an over-ride switch which allows the operation of the emergency outdoor air intake dampers in the presence of a control room isolation signal.

Despite the fact that the possibility of fire in the control room is low, provisions have been made to prevent recirculation of smoke-filled air to the control room. Smoke detectors for the MCR, relay and switchgear rooms, Control Rod Drive Mechanism (CRDM) control room, and cable spreading areas will alarm the control room and the operator may shift the outside air intake and the smoke relief dampers to open fully and signal the return air damper to close fully, thereby causing all air to be exhausted to the atmosphere.

The MCR-HVAC is designed as a redundant, safety, seismically qualified system which is energized (each train) from independent safety power bus. In the case of loss of ultimate heat sink the system could work without the limitation (MCR chiller system) due to the Air cooled chiller water system. In the case of Loss of All AC power to the plant (SBO) manual operation will be required to energize the minimum capacity of MCR-HVAC to keep the living conditions in the MCR adequate. The effect of loss of the ventilation in NPP MCR has been evaluated to be acceptable providing action is taken to enhance passive equipment cooling when needed and the temperature does not exceed 49 °C. In this case also limited heat energy sources would be available in the MCR and consequentially the heating of the MCR would be reduced.

In case air quality in the MCR worsens and operators cannot breathe normally, 12 sets of breathing apparatus with 24 tanks with compressed air are available at all times just outside the control room. One tank can be used for about one hour for breathing. More tanks are available onsite, together with diesel-powered charging compressor.

If the MCR has to be evacuated due to any reason, there are 3 evacuation panels available in the plant with sufficient control and monitoring capability to safety cooldown the plant to the cold shutdown under the direction of the special set of operating procedures.

Two different scenarios concerning the evacuation of the MCR are covered by two different procedures. If the MCR have to be evacuated due to the smoke or chloride in the MCR the procedure to stabilize the plant from the remote shutdown panel will be used. This procedure basic assumption is that the most of the equipment (circuitries, switches, etc.) will be operable all the time, but the MCR is inaccessible. The operators will occupy their temporally position on the different shutdown panels and will establish the control over all shutdown systems and functions. Operators will be coordinated by the Shift Supervisor who will conduct the operation from the TSC.

In the case of the major inoperability of the equipment in the MCR and consequentially disable capability of the operation of the plant from the MCR (basically the fire in the MCR or in the cable spreading room are identified as a major potential cause for this event) the operators will be evacuated from the MCR after the declaration of the MCR evacuation by the Shift Supervisor. There are some immediate actions steps defined in this fire response procedure given the instructions to shutdown the reactor, turbine, etc. to put the plant in the stable state. After the evacuation of the MCR the operators will be sent to the same positions on the shutdown panels, however due to the potential problems, inadvertent operation and equipment malfunctions more local operation have to be implemented to stabilize the plant. Stabilization and cooldown of the plant to the cold shutdown shall be then implemented by this set of procedures.

6.1.2.2.2 Impact on the different premises used by the crisis teams or for which access would be necessary for management of the accident

Krško plant performed a post-accident shielding review in 1994. The plant examined the actions which could be taken to reduce high radiation and increase the capability of operators to control and mitigate the consequences of the accident.

The dose rates as calculated were extremely conservative, based on conservatism in the prescribed source term and conservative instantaneous release from the core. The access to some locations with the presence of post-accident coolant is made only if actual dose rates allow it.

Safeguard features are designed to operate for one year following the accident and in-service inspection and maintenance ensure pump operability and reliability. In case of any local actions, particular care shall be taken if and when access to such a room is required.

Generally, pump rooms were found to be inaccessible once the pump was in operation with post-accident coolant, at least for the short to medium term. The presence of post-accident coolant in any single train is generally enough to restrict access severely. However for the most of the locations on the plant following the DBA the dose rate were found to be acceptable, considering the required access. As the exception the penetration areas in auxiliary building where identified, where short term access, for whatever duration will be impossible under the present assumptions. Access to these penetrations areas has been required, in the short term, for surveillance of valve positions and in extreme case for manual intervention in case of failure. It shall be noted that redundancy in valve arrangement is provided and that plant preventive maintenance will ensure valve operability. Remote position indication is provided as well.

Short-term access to pump rooms is required only for surveillance, except for the CVCS positive displacement pump (PDP). Short-term surveillance is not deemed to be essential, in view of the availability of remote indication of pump operation and the in-service inspection and maintenance ensuring pump operability. Therefore, these actions can be performed if actual dose rates permit them.

The plant is capable of monitoring high dose rates up to 100 or 1000 mSv/h with the installed area radiation monitors at various locations. The corridors are equipped with emergency lighting with permanent battery power supply. Any entry to critical locations has to be reviewed in advance and shift radiation protection technician should be capable of measuring and assessing radiation conditions. In any case, when an access to the areas with a dose rate exceeding 10 mSv/h is foreseen, it needs to be specifically approved.

<u>Future plans to improve the access for accident management</u>

The plant has triggered additional actions for further evaluation of the post-accident shielding review and to install in the near future some simple shielding arrangements on the piping near the valves which might be accessed during the accident.

In this context it may be mentioned that a simple shield arrangement may facilitate access to a few locations or rooms considerably. This would reduce the restrictions on operator presence and/or the dose expenditure for certain post-accident actions and deserve consideration for this reason.

There is also a plan to install post-accident area radiation monitors with battery power supply and radio link within the corridors close to the piping and valves in question which might be accessed or even should be accessed for the reason of post-accident sampling.

Radiation protection technicians are going to be equipped with audio links and sufficient wireless communication channels. Central radio communication equipment and personal electronic dosimetry performance is going to be evaluated and improved if necessary as well as related power supplies.

6.1.2.3 Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods)

For the implementation of Krško NPP's RERP the following emergency response (in addition to the MCR) are established:

- Technical support centre (TSC),
- Operational support centre (OSC)
- Emergency operations facility in Ljubljana (EOF).

The TSC is recognized as a coordinating center of operation, maintenance and other activities during an emergency. The MCR is the main center for the physical control over the plant systems and components. Even in the case of the TSC operability, all physical interactions and controls would be initiated via the MCR.

Physical diversity between the TSC and OSC should be noted. TSC is situated on the first floor of the health physics building and the OSC is located in the underground concrete shelter. Both centers are equipped with different habitability systems therefore ensuring the access to them during any postulated external event (earthquake or flood).

In the case of an inaccessibility of TSC or/and OSC, there are procedural instructions when and where the TSC or OSC staff should be evacuated. As the first back-up location for the TSC the MCR could be used. In the case of the plant inaccessibility, the alternative location in the Krško city should be used. For the plant emergency staff, the Krško location is prepared and equipped with basic communication equipment and documentation needed for the implementation of TSC/OSC functions.

TSC lighting and HVAC are energized from redundant emergency power supply buses, however, there is an alternative to supply TSC from alternative diesel generators under the manual breaker operation.

6.1.2.4 Unavailability of power supply

The Krško NPP is connected to the network of Elektro-Slovenija, d.o.o., Slovenia, via the 400 kV transmission system terminated at the 400 kV switchyard. The separate 110 kV transmission line from Krško or Brestanica distribution and transformer station is terminated in a bay where 110 kV backup source is provided. The emergency power source for the plant is onsite, independent power source which consists of two diesel generator units and the DC battery systems, designed with sufficient capacity to furnish onsite power to reliably shut down the reactor, remove reactor residual heat, supply control and instrumentation power, monitor essential reactor parameters, initiate operation of protective equipment and reactor building isolation, when required. Reliability is assured by the use of independent controls and sources to supply AC and DC engineered safety feature loads. The emergency power supply ensures that the plant can be shut down and maintained in a hot shutdown condition without loss of the engineered safety features described in the Class 1E power system above.

In the case of SBO, the plant can be stabilized and cooled with the independent AFW TDP. DC systems supply power for circuit breakers control and vital instrumentation control. Upon total loss of AC power, the batteries supply uninterruptible electrical power to the DC systems until either offsite power is restored or onsite power from emergency diesels is available. Critical 118 V AC instrumentation and control is powered from the DC system through inverters to provide a reliable and transient free power supply. This design provides continuous monitoring and control of critical instrument channels (instrumentation operation under the degraded site power supply is described in the next Section 6.1.2.5).

Three separate and independent DC battery systems are provided for the unit, two 125 V DC and one 220 V DC. Each 125 V DC system consists of a battery, a battery charger, a main distribution switchboard with air circuit breakers, local distribution panels, feeders and associated equipment. Batteries, chargers and distribution systems are located in separate locked rooms in a Seismic Category I structure. The system is sized to provide DC power under LOCA conditions. Adequate capacity is available during simultaneous loss of AC power and subsequent safe unit shutdown. The batteries have sufficient capacity to cope with a 4-hour station blackout (loss of all AC power), to provide a safe plant shutdown. It is considered that starting at least one emergency diesel generator within that period can be achieved. However, the usage of the DC batteries could be prolonged to 16 hours and above (see 5.1.2.1.1) by stripping additional equipment from the battery supply according to emergency operating procedures.

Each Class 1E train is provided with a complete 125 V DC system which supplies DC power to loads associated with the train. Each train's system consists of a full capacity 125 V DC lead-acid 60 cell battery, 125 V DC switchboard, solid state battery charger and required distribution boards. The battery charger is arranged to supply the DC system and to provide the float charge to the battery during normal operation. Upon loss of station AC power, the entire DC load is supplied by the battery. The important instrumentation and control is powered from 118 V distribution which is powered through inverters powered from DC system.

The battery charger is sized to carry normal plant operation DC loads while recharging a fully discharged battery in 12 hours. Each train has access to an installed swing charger which in turn can be fed from its associated train 400 V AC source. Interlocks are provided to ensure separation of the redundant trains.

The batteries are sized to supply DC loads as defined above for a minimum of four hours with a final discharge of 108 V (1.80 V per cell). The batteries have sufficient capacity per design to cope with a 4 hour station blackout (loss of all AC power), to provide safe shutdown of the unit. The capacity of each battery is 2080 Ah.

EOPs instruct the operators to disconnect all non-essential DC loads. Based on plant specific best estimate DC study and with the actions of the operating crew to disconnect all non-essential DC loads, the above mentioned 4 hours will be extended to and above 16 hours (train A) and 13.5 hours (train B). However with the multiplication of additional diesel generators (one fixed and five mobile), the instruction to strip all non-essential DC loads loses priority as the diesel generators ensure much longer availability of the buses. Establishing alternative power supply to the bus LD11 and to battery chargers from one of the two portable diesel generators will assure the long time availability of DC batteries and of 118 V AC instrumentation power supply (up to 72 hours since fuel is stored at the plant for this time period, or even longer if fuel would be supplied form outside of the plant).

The plant is further equipped with a non safety grade 220 V DC system sized to provide power to Turbine emergency oil pump, Emergency seal oil pump, Lighting panel, DC panel for control of 400 kV substation breaker and inverters for PIS in the event of AC power failure. The 220 V battery is sized to supply the above mentioned DC loads for a minimum of four hours. The battery has a final discharge voltage of 1.80 V per cell. The capacity of the battery is 2175 Ah.

In case all of the above plant design AC power supply fails, two auxiliary rugged container diesel generators are stored on-site, with 600 kVA and 1000 kVA rated power each, and either being able to supply 400 kVA to the 400 V safety bus (due to transmission cable limitations). Location of the generators is approximately 150 m away from and 2 m above the plant's safety-class diesel generators, thus making them unsusceptible to the common cause failure of the safety-class diesel generators. Both container diesel generators use jacket water heater and their batteries are constantly filled with charger. They are also capable of cold-starting at -20 °C. Enough fuel is stored onsite for diesel generators' 3-day operation. It takes less than 1 hour for operating crew to connect the selected container diesel generator and start delivering electrical power to the 400 V safety bus.

The primary purpose of the container diesel generators is to power battery charger, which then provides power to one train of plant's AC and DC control power, thus enabling control room indications and controls, as well as control room lighting. Also, self-cooling positive displacement charging pump can be started, providing limited core injection capability. Depending on the momentary load of the transmission cable, there is some power left available to manipulate motor-operated valves as needed.

In addition to the two container diesel generators, three 150 kVA diesel generators are placed around the nuclear island. Their operating location is on the yard, close to the motor control centers they are intended to supply. In case of unavailability of any other AC sources, these generators can be connected to and can power their respective motor control centers by simply plugging the ready cable into the socket, which is mounted on the wall near the motor control centers. The 150 kV generators are intended to provide quick alternative power supply to motor-operated valves, thus enabling plant's staff to manipulate them as needed.

Water pumping capabilities on the plant are backed-up with two mobile gasoline-powered firefighting pumps, which are stored onsite. With provisions on the systems with universal firefighting connections installed, they can pump water from virtually any tank onsite, and discharge water into the systems as directed by TSC in order to respond to containment challenges. Additionally, trailer-mounted diesel-powered submersible pump is stored onsite, which can be easily deployed to pump the Sava river water to the plant systems. These pumps run free of any of the plant systems, and can thus be successfully used even in the event of a prolonged station blackout.

In case of unavailability of the AC power and/or cooling water for the instrument air compressors, two diesel-powered high-capacity trailer-mounted compressors are stored onsite. Provisions on the instrument air system with quick-connectors allow them to supply compressed air to the IA system, both inside and outside of containment, even in case of prolonged station blackout and/or loss of ultimate heat sink. Compressed air can be used to manipulate air-operated valves in order to achieve important goals in protecting containment integrity, such as depressurize the RCS or open containment vent path.

The **Normal Lighting System** provides plant lighting under normal operating conditions. It consists of incandescent, fluorescent, and high intensity discharge light sources operating at 220 V AC, and is fed from 400/230 V, 3-phase, 4-wire normal lighting panels. These panels are fed from non-Class 1E 400 V buses. Lighting in the reactor building, auxiliary building, fuel handling building, component cooling building, and personnel airlock area is accomplished exclusively with incandescent light sources. On the other hand the **Essential Lighting System** operates in conjunction with the normal lighting, and is utilized in those areas where highly reliable illumination is required for safe access or egress, or the combination of critical tasks. The system operates at 230 V AC, fed from 400/230 V, 3-phase, 4-wire essential lighting panels. These panels are fed from 400 V safety feature buses. In areas designated as Train A areas, only Train A essential lighting is provided. Similarly, in areas designated as Train B areas, only Train B essential lighting is provided. Essential lighting is provided for safe passage in the auxiliary building, diesel generator building, intermediate building, fuel handling building, control complex, component cooling building and essential service water pump-house. It is also provided for continuation of critical activities at the diesel control panels, DC distribution panels, inverter and charger areas, switchgear, relay and computer rooms and parts of the controlled access areas.

In the MCR, all lighting is essential and is divided equally between Train A and Train B. In this manner, failure of either train will only affect 50% of the control room lighting. These areas are also provided with additional emergency lighting units with double lamps and 8-(eight)-hour battery backup.

In the case of emergency (Loss of All AC) the emergency lighting system is provided for purposes of egress in all contiguous plant areas where failure of the normal and/or essential lighting systems may hamper safe personnel egress. In the MCR, parts of the controlled access area, and diesel generator rooms, emergency lighting is provided in sufficient quantity to ensure continuance of critical activities upon loss of all other light sources. Emergency lighting is supplied from the 220 V DC battery. In addition, the **Additional Emergency Lighting System (AEL)** provides backup illumination in all areas needed for operation of safe shutdown equipment and access and egress routes thereto. AEL units are arc self contained, sealed-beam, double lamps units equipped with integral battery packs rated at 8 hour min. Units are powered from existing essential lighting panels. In a case of blackout (loss of input line voltage), units are automatically turned on, and after restoration of AC power units are automatically turned off.

6.1.2.5 Potential failure of instrumentation

The safety related instrumentation systems are designed to meet the independence and separation requirements of IEEE 279-1971 Criteria for Protection Systems for Nuclear Power Generating Stations. The electrical power supply, instrumentation, and control conductors for redundant circuits of a nuclear plant have physical separation to preserve the redundancy and to ensure that no single credible event will prevent operation of the associated function due to electrical conductor damage. Critical circuits and functions include power, control and analog instrumentation associated with the operation of the Reactor Trip System (RTS) or Engineered Safety Features Actuation System (ESFAS). Credible events include, but are not limited to, the effects of short circuits, pipe rupture, missiles, etc. and are considered in the basic plant design.

Instrumentation channel independence is carried throughout the system, extending from the sensor through to the devices actuating the protective function. Physical separation is used to achieve separation of redundant transmitters. Separation of wiring is achieved using separate wire-ways, cable trays, conduit runs and containment penetrations for each redundant channel set. Redundant analog equipment is separated by locating modules in different protection rack sets. Each redundant channel set is energized from a separate AC power feed. There are four separate process analog rack sets. Separation of redundant analog channels begins at the process sensors and is maintained in the field wiring, containment penetrations and analog protection racks to the redundant trains in the logic racks. Redundant analog channels are separated by locating modules in different rack sets. The separation criteria presented also apply to the power supplies for the load centers and busses distributing power to redundant components and to the control of these power supplies.

In the case of instrumentation failures the emergency operating procedures and severe accident guidelines give instructions to use redundant and alternative indications. Redundancy, separation and diversification of the instrumentation give sufficient insurance that at least the minimum instrumentation will be available during the postulated accidents. The most serious postulated case regarding the availability of the instrumentation is therefore the availability of the power supply to the instrument buses. As mentioned in Chapter 6.1.2.4 there are alternative means and methods available to enable the operation of at least limited instrumentation channels.

6.1.2.6 Potential effects from the other neighboring installations at site

A few industrial and transportation facilities are located near the site. The existence of these facilities does not affect the safe plant operation. The Zagreb Airport, located approximately 50 km south-east of the site, has a 5.000 m long paved runway. The flight paths do not pass over the site. All commercial, heavy and private, light aircraft traffic is controlled within a 150 km and 30 km radius of the airport, respectively.

The Ljubljana Jože Pučnik Airport, located approximately 80 km north-west of the site, has a 3.000 m long paved runway. All flights into and out of Ljubljana Jože Pučnik Airport are controlled by the airport tower on regulated flight paths. There is also a smaller military airport Cerklje, located 4.3 km to the south from the site, which is also used for civil purposes. The impacts of all three airports are briefly described in Section 2.1.2.5.3.

The effects of explosions or fires due to nearby railroad or road accidents are negligible on Class 1 structure buildings. In addition, plant Fire protection system is adequate. There are also some factories located in the plant vicinity. No safety measures are required since there are no hazardous impacts of potential accidents in Paper plant VIPAP Krško or any other nearby industrial facility.

Krško NPP is a single unit plant; however, there are some hazardous materials storage tanks located onsite. They are dislocated from the technological part of the plant. These locations (underground emergency diesel generators fuel tanks, auxiliary boilers fuel tank, plant hydrogen etc.) are marked in the plant Fire protection plan which lists the Fire protection fighting equipment, evacuation plans and other useful information for the fire fighting brigade in the case of the fire onsite. The locations with dangerous material are dislocated to provide physical separation between the technological part of the plant and these storage locations.

The Fire protection and Defensive plan and Fire response procedures provide instructions on how to manage the potential fire situation and what are the measures which have to be taken in case of fire at the plant. Each location (area) at the plant (also for non-technological part) is covered with the special drawing presenting all the elements useful for the fire fighters in the case of the fire on the plant. In the case of the fire in the technological part of the plant, a special set of procedures is prepared in accordance with the USNRC 10CFR Appendix R which gives the instructions on how to achieve safe plant shutdown and further plant cooldown to the cold shutdown. These sets of procedures have been prepared in accordance with the Appendix R Fire Hazard Analyses and Safe Shutdown.

Generator hydrogen release and potential fire in the turbine building is one of the postulated accidents following a loss of SBO or LOOP. In this case the EOPs give the instructions on how to actuate the emergency release of the hydrogen from the generator which can start to leak due to the potential loss of Generator hydrogen sealing system. The generator hydrogen is released via the release path on the turbine building roof equipped with the hydrogen burn arrestors. Fire protection fire fighting systems are designed to stay operable even in the case of the SBO by using diesel fire protection pump.

The fire protection system is designed for prevention fire hazards. The fire protection system cannot prevent a fire from occurring, but does provide the facilities for detecting and extinguishing fires in order to limit the damage caused by a single fire.

In addition to the fire protection system itself, there are many design features of the plant which would also contribute to confining and limiting a fire condition. The building structures are constructed of fire resistive concrete. The power plant is divided into several buildings that are separated from each other by fire walls. These buildings are: reactor building, auxiliary building, control building, fuel handling building, intermediate building, diesel generator building, turbine building, and component cooling building. In addition, stair towers, in all but the reactor building, are enclosed with fire rated walls. The two emergency diesel generators are separated from each other by a two hour fire rated wall. The turbine lube oil reservoir and lube oil conditioning equipment are in a room which is separated from other areas of the turbine building by two hour fire rated construction. The combustible material in each particular fire zone is well controlled and has been inventoried. The standard calculation has been used within the actual fire hazard analysis to calculate the combustible load of the particular fire zone. The combustible load (time of burning) of the particular fire zone is lower that the fire rating of the separation features (walls, doors, penetrations etc.).

Extensive vertical runs of cables, ducts, and pipes are either enclosed in shafts with all shaft openings sealed with a noncombustible fire rated material, or all openings around cables, ducts, and pipes passing through major floors are sealed with a noncombustible fire rated material.

Large oil filled transformers are located outdoors so that a fire would not damage the plant buildings. In addition fire barrier walls are located between the individual transformers and between the transformers and any air louvers in the walls of the turbine building. This limits a fire condition to only a single transformer without affecting the turbine building interior or an adjacent transformer.

The fire protection water distribution system consists of outdoor underground piping and yard fire hydrants and interior fire protection distribution and standpipe system. Water is pumped into the outdoor underground yard piping by the fire pumps. The outdoor yard piping is arranged in a loop with several connections to the plant buildings to supply all fire protection water for the fire protection within the buildings.

6.2 Nuclear power plant

6.2.1 Accident management measures currently in place at the various stages of a severe accident, in particular subsequent to a loss of the core cooling function

6.2.1.1 Before fuel damage in the reactor pressure vessel

The main safety objective in reactor plant design and operation is control of reactor fission products. The methods used to achieve this objective are:

1. Fuel protection: reactor core design in conjunction with reactor control and protection systems to preclude the release of fission products from the fuel.
2. RCS Integrity: retention of fission products in the reactor coolant for whatever leakage occurs.
3. Containment integrity: retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary.
4. Environment: limiting or optimizing fission product dispersal to minimize population exposure for an accidental release beyond containment (accomplished by imposing operational limits e.g. RCS activity).

Engineered safety features (ESF) is the designation given to systems provided to protect the public and plant personnel by minimizing both the extent and the effects of any accidental release of radioactive fission products from the reactor coolant system, particularly those following a LOCA. These safety features function to localize, control, mitigate, and terminate such accidents and to hold the offsite environmental exposure levels within the limits.

This concept of ESF is used in the design of safety-related systems which directly mitigate the consequences of a design basis accident (DBA). A DBA is a postulated accident that a nuclear facility must be designed and built for to withstand without loss to the systems, structures, and components necessary to assure public health and safety.

ESFs have been designated to provide protection during any size break and type of a reactor coolant pipe (assuming unobstructed discharge from both ends), and any steam or feedwater line break.

The following SSCs are provided to satisfy the above cited functions and are designated as ESFs:

- Containment systems
- Emergency core cooling system (ECCS)
- Control room habitability system
- Reactor building annulus negative pressure control system

The containment systems consist of the steel shell containment, concrete shield building, penetrations, and the directly associated systems upon which the containment functions depend, as follows:

- The containment isolation system isolates the various fluid systems that pass through the containment wall to prevent the direct release of radioactivity to the environment in the event of a postulated accident.
- The containment spray system has a dual function of containment atmosphere heat removal by spraying of borated water through the containment; and to enhance fission product removal efficiency and prevent significant re-evolution of the dissolved iodine species back into the containment atmosphere as volatile after the recirculation phase begins.
- The containment air recirculation and cooling system maintains the containment atmosphere at or below the design pressure and temperature by transferring the containment heat to the component cooling water system. This serves to reduce the leakage of airborne radioactivity from the containment building following an accident.
- A combustible gas control system is provided for the post-accident control of hydrogen in the containment. This is accomplished by processing the containment air through the electric hydrogen recombiners.

See also paragraph 6.1.1.5.

The emergency core cooling system (ECCS) ensures the delivery of a timely, continuous and adequate supply of borated water to the reactor coolant system. This provides core cooling to limit fuel cladding temperature and fission product release, and ensures adequate shutdown margin. The system also provides continuous long-term, post accident cooling of the core by recirculation of borated water from the containment sump. Core cooling is provided immediately following a LOCA by accumulator injection, safety injection and residual heat removal pumps, and their associated valves, tanks and piping. After injection, water collected in the containment sump is cooled and returned to the reactor coolant system via the emergency core cooling recirculation paths.

The ECCS shall be designed such that its cooling performance following a postulated LOCA conforms to the following criteria:

- Peak cladding temperature will not exceed 1200 °C.
- Cladding oxidation will not exceed 17% of the total cladding thickness.
- Hydrogen generation (due to zirconium-water reaction) will not exceed 1% of the hydrogen generated if all the zirconium surrounding the fuel reacted.
- Core remains in a coolable geometry.
- Long-term cooling capability will be maintained (i.e., core temperatures remain acceptably low and decay heat is removed for the time ECCS operation is required).

Another design criterion is reliability. The ECCS reflects this in several ways, one of which is its failure modes. The ECCS is designed to accept a single active failure following an accident (with a loss of site power) without loss of its protective function. It is also designed to accept a single active or passive failure during the recirculation mode. An active failure is defined as the failure of the component (i.e., valve, pump, etc.) to operate. A passive failure is defined as the failure of a passive component (i.e., valve packing leakage, flange break, etc.).

The control room habitability systems allow the plant operators to safely occupy the control room for an extended period in order to maintain the nuclear power plant in a safe state under postulated post accident conditions.

The reactor building annulus negative pressure control system collects the leakage from the reactor containment into the annulus between the reactor containment vessel and the shield building, and discharges it through filters to the plant vent.

Reliability of ESF lies in design philosophy which was taking into account systems redundancy (safety system often consist of a number of individual, functionally identical system known as trains), diversity of components (for the same safety function there are more different physical methods used for achieving it), separation (physical and electrical), automatic response, ability to test and inspect while reactor is in operation, single active failure (a failure of a component necessary for system safety will not prevent system from achieving design purpose), safe failure of a system or a component to a position necessary for plant accident condition, certification (safety components are suitable for harsh conditions – temperature, humidity, radiation, etc.).

General design criteria cover protection by multiple fission product barriers, protection and reactivity control systems, fluid systems, containment design, and fuel and radioactivity control. Defense in depth concept contains several levels of protection including successive barriers preventing the release of radioactive material to the environment. The levels of protection in defense-in-depth are:

1. a conservative design, quality assurance, and safety culture,
2. control of abnormal operation and detection of failures,
3. safety and protection systems,
4. accident management, including containment protection; and
5. emergency preparedness.

6.2.1.1.1 Preventive measures

Given the existence of the automatic ESFs, the EOPs, and well-trained licensed operators, the probability that any initiating event will lead to core damage is low. However Krško NPP does not consider it negligible, therefore a severe accident management program has been implemented in addition to general, abnormal and emergency operating procedures development. It includes both the development of plant-specific severe-accident management guidelines and training of personnel who would be tasked with managing a severe accident, should one ever occur.

According to the reactor plant design and operation NPP has developed procedures which are responding to any abnormalities of the particular system. MCR operating staff has been trained to respond by the plant condition by appropriate sets of procedures: Alarm Response Procedure (ARP), Abnormal Operating Procedures (AOP), Emergency Operating Procedures (EOP), Function Restoration Guidelines (FRG) and Fire Response Procedures (FRP). Figure 14 is showing operators response to abnormal situation by using different level of procedures.

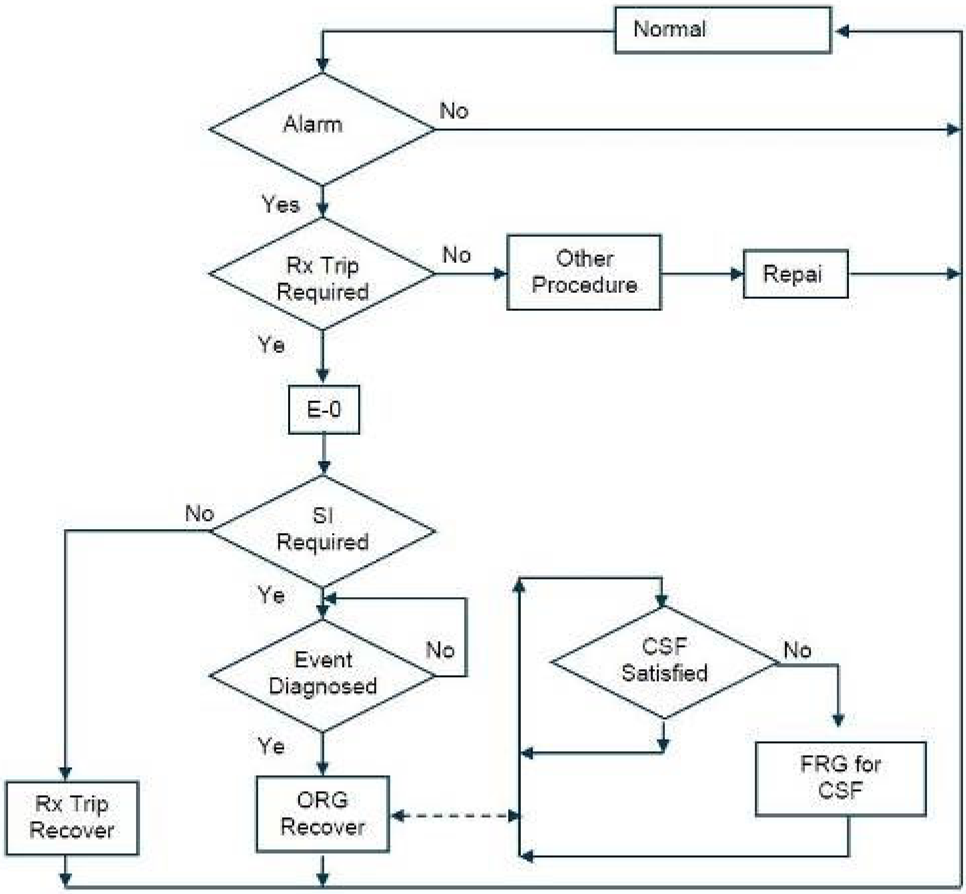


Figure 14: Emergency response guideline usage

Alarm Response Procedures (ARP) guide operators to take proper action in their response to MCR alarm conditions. If significant plant parameter or deviation exists, ARP direct control room operating staff to appropriate AOP or EOP. For instance, if seismic event is occurred the MCR alarm annunciator directs MCR staff at appropriate AOP or EOP, depending of event severity.

Abnormal Operating Procedures (AOP) are intended to handle abnormal occurrences with no reactor trip or to support mitigation of the occurrence when reactor trip occurs. The procedures cover the primary systems malfunctions or abnormalities, reactivity control abnormalities, secondary systems malfunctions or abnormalities, refueling conditions abnormalities, instrumentation systems inoperability, electrical systems malfunctions, radiation abnormalities and explosive mixture occurrence, cooling capabilities malfunctions and environmentally induced abnormalities (e.g. earthquake and flooding).

If significant plant parameter or deviation exists, it directs control room operating staff to appropriate AOP or EOP.

Emergency Operating Procedures (EOP) are written so that trained operational shift crew will be able to identify an emergency from the symptoms available, take immediate actions on the expected course of the event, mitigate the consequences, and place the plant in a stable and safe condition. These procedures are entered always when reactor trip occurs or when it should occur. In the EOPs, the emphasis is on preventing core damage. The EOPs contain two types of procedure, whose use depends on whether the event can be diagnosed or not (Figure 14).

Optimal Recovery Procedures deal with situations where diagnosis is possible, and they cover both design basis situations such as:

- Reactor trip with or without a safety injection,
- Loss of coolant (primary or secondary),
- Steam line break,
- Steam generator tube rupture

and also certain beyond design basis situations such as for example:

- Loss of all AC power
- Loss of primary coolant recirculation
- Uncontrolled depressurization of all SGs
- And numerous others.

For situations where diagnosis is not possible, **Function Restoration Guidelines** (FRG's) are provided. FRGs provide an explicit, systematic mechanism for evaluation and restoration of the plant safety state in terms of Critical Safety Functions (CSF) status As long as the fuel matrix/cladding, RCS pressure boundary and containment barrier are intact, the plant poses no threat to the health and safety of the public. CSFs, which are continuously monitored after entry to EOPs, if satisfied, are sufficient to maintain the fuel matrix/cladding, RCS pressure boundary and containment vessel barrier. CSFs in order of priority are as follows:

1. Subcriticality (minimizing energy production in the fuel),
2. Core cooling (providing adequate reactor coolant for heat removal from the fuel),
3. Heat sink (providing adequate secondary coolant for heat removal from the fuel),
4. RCS integrity (preventing failure of RCS),
5. Containment integrity (preventing failure of containment vessel),
6. Reactor coolant inventory (providing adequate inventory).

Relation between CSFs and barriers is shown on Figure 15.

It is important to note that the EOP package deals with preventive measures for all types of event: those within the design basis and also those beyond design basis.

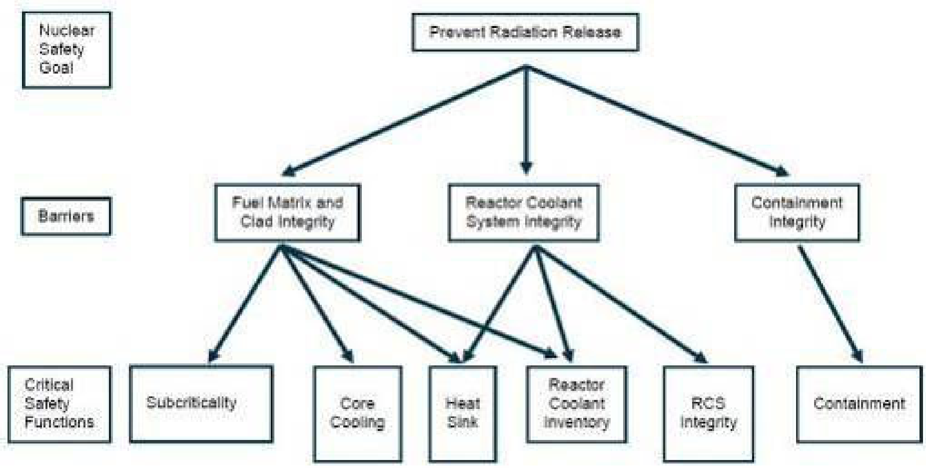


Figure 15: Relation between FRG's goal, barriers and CSFs

Fire Response Procedures (FRP) address operator response to the fire in the technical part of the plant. The basic intent of this set of procedures is to achieve safe shutdown after the initiation of fire in any of the fire zones. The design basis fire event assumes also the loss of all off-site electrical power supply. The FRP procedures are used when the presence of fire has been confirmed. FRPs are based on the mitigation of the consequences caused by fire impact on the safety equipment. If a fire takes place in any other places than the MCR or cable spreading room below the MCR, the FRPs would be used in parallel with AOPs or EOPs to stabilize the plant and achieve safe plant shutdown. In the case of fire in the MCR, a specific set of procedures shall be used to mitigate the consequences even from the outside of the MCR, if evacuation is required. This set of procedures also includes the procedure for the operation response in the case of the loss of the component cooling or service water system.

6.2.1.1.2 Kinetics for entering a severe accident, cliff edge effects

A combination of substantial equipment failure and consequently a lack of operators' actions is the most likely scenario that leads to the core damage event. For core damage to occur the core must be uncovered and remain uncovered long enough to overheat. Initially, core heat-up is driven by the decay heat. But once the cladding becomes hot enough, the heat released by the zirconium-steam reaction dominates and accelerates the core heat-up.

The various types of operating procedure and guideline and their interfaces, are shown in Figure 16.

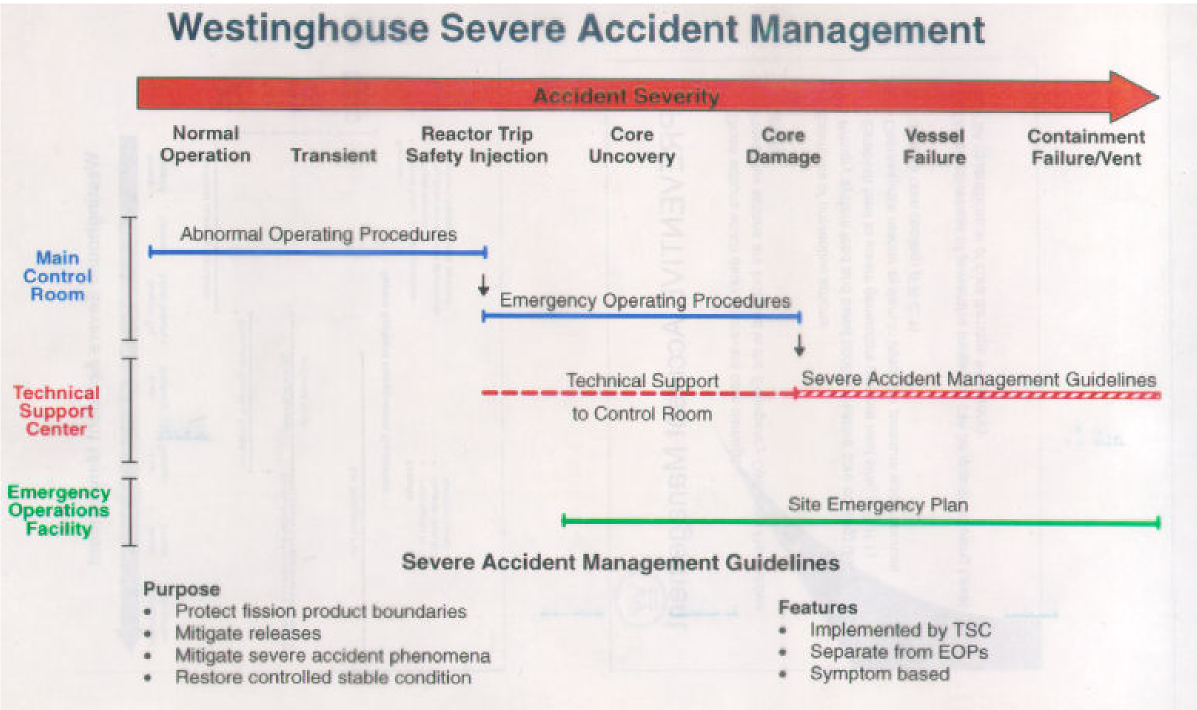


Figure 16: Graphic structure of Krško NPP procedures and guidelines in case accident scenario is leading to core damage

Krško NPP performed analyses where long-term SBO accident sequences were calculated with the focus on the containment response after the core damage. Accordingly, several scenarios were evaluated with respect to different RCP seal leakage flow, availability of AFW TDP, possibility of gravity feed from the RWST to the containment sump and the alternative spray/injection using the alternative sources.

Following the SBO initiation, operators would act per »Loss of all AC power« EOP. Reactor coolant pumps (RCP) seals would lose their cooling and it is assumed that the coolant will be discharged through the seals from the beginning of the transient. The seal leakage rate applied is in accordance with plant data that are using a high temperature o-ring RCP seal packages. If emergency diesel generators fail to start, and auxiliary offsite power sources are not available, operators immediately start establishing an alternative power supply from one of two available mobile diesel generators (located onsite with 3 days fuel supply) for running charging positive displacement pump (PDP to supply water to RCS) and charging the safety batteries to provide instrument and control power. In addition, operators would isolate letdown line by using portable diesel generators for establishing power to motor operated valves inside containment.

If there were no actions taken for isolating letdown line, this would consequently lead to the opening of letdown relief valve to pressurized relief tank (PRT) increasing the coolant loss until RCS pressure decreases below valve set-point. If assumed that PDP was not in operation (by any reason), reactor coolant losses would not be replaced. Operators would rapidly depressurize secondary side of SGs using PORVs, and this would lead to the primary side cooldown and depressurization. RCS pressure decrease results in passive accumulator injection to RCS, which lasts until RCS pressure drops below pressure when accumulators should be isolated. AFW TDP does not require electric power and can operate if the SG pressure is appropriate, so it can provide secondary injection for infinite time. Taking into account the assumption that the AF flow control valves are operable (4 hours nitrogen or alternative compressed air supply or manual operation) and that condensate storage tanks (CST) can be refilled by variety of sources (demineralized water storage tanks, pretreated water tanks, fire protection tanks, condenser hotwell, city water, circulating water tunnel, Sava river water), the secondary side heat sink would be available during the whole transient. Even if AFW TDP is not operable, Krško NPP has available portable fire protection pumps with fuel supply for 3 days of operation onsite, with variety of injection flow paths to SGs and variety of water sources available. Fire track can also be used. The primary coolant loss is rather low and rapidly decreases when the RCS water level falls below leakage elevation through RCP seals. RCP seals are also cooled by cooling of the RCS. Consequently, the core does not uncover within 7 days, therefore the core, reactor coolant system and the containment integrity are not endangered.

By providing whichever sources of heat sink (main feedwater, auxiliary feedwater or alternative by using portable fire protection pump) and assuring its long-term availability (condensate water or alternative source), core damage can be prevented and progression of an accident to the containment can be stopped.

With complete loss of heat sink, core damage would occur if no means for water injection into RCS was available for decay heat removal and replacing inventory loss by feed and bleed method. Feed and bleed method can be established by injecting water into depressurized RCS also by portable fire protection pump taking suction from variety of borated or un-borated water sources.

The most important systems with respect to core damage prevention are AFW and emergency diesel generators. This is in accordance with the high importance of secondary heat sink function and with the high contribution of LOOP and SBO initiators to core damage probability.

Therefore Krško NPP has a variety of alternative methods for injecting water into both SGs in the case of SBO together with mobile diesel generators for providing electrical power to particular equipment (battery charger, PDP, particular MOVs). This mobile equipment is placed onsite and has fuel available for three days of operations onsite as well. When AC power is restored, main feedwater pumps, condensate pumps, service water pumps, electrical fire protection water pump can be used.

Necessary operators’ actions may also be ranked using the same measures of importance. Among the most important operators’ actions to prevent core damage are establishment and/or verification of secondary heat sink and establishment of primary feed and bleed water injection into the RCS upon failure of the secondary heat sink.

Complete loss of heat sink and no means for water injection into RCS for decay heat removal and replacing inventory loss will cause core uncovery and heatup, and eventually cause the core melting. Therefore core damage is recognized as a cliff edge effect. Detailed description is given in section 6.2.1.2.1.

6.2.1.1.3 Possible actions for preventing fuel damage

Before fuel damage occurs, operators in the MCR will be guided through EOPs and FRGs to three particular procedures, which respond to specific accidental plant conditions. They are based on analyses of transients which could lead the plant to a severe accident in which core damage is expected. Therefore they would provide actions, which would maintain the integrity of the core material and prevent fission product release. These procedures are:

1. Loss of all AC power,
2. Response to inadequate core cooling, and
3. Response to nuclear power generation (Anticipated transient without scram (ATWS) event).

<u>Loss of all AC power (station blackout - SBO)</u>

A SBO at a nuclear power plant can result only from a coincidence of loss of power from the high voltage grid transmission lines and a combination of events preventing the station emergency diesel generators from energizing the emergency AC busses. The immediate consequences of the loss of AC power, if not accompanied by another accident such as a loss of reactor coolant, loss of secondary coolant or steam generator tube rupture, are not severe. However, should AC power either from the grid or the emergency DG not be restored in time, the consequences to the plant could become severe. The degree of severity of a SBO depends primarily on the duration of AC power outage and the response of a RCP shaft seals to the loss of seal cooling (i.e. concurrent loss of injection flow to the RCP seals and component cooling flow to the RCP thermal barrier). Without power this leakage cannot be replaced and a continuous loss of reactor coolant occurs with time. Also letdown flow must be isolated. To mitigate severity of SBO, it is necessary to minimize RCS inventory loss, and to restore AC power. Consequently, any action to reduce RCS pressure and temperature during a SBO event is consistent with minimizing RCS inventory loss and assuring adequate decay heat removal which will maximize time to core uncover. Krško NPP possesses alternative equipment stored onsite. In addition to two redundant safety-related diesel generators, which are providing power to two redundant and independent trains of engineered safety systems, this equipment includes onsite mobile diesel generators for establishing alternative power supply to pump for RCS injection, some critical motor operated valves, battery charger (battery) and instrument buses. Alternative equipment is supplied with fuel and logistics for three days of continuous operation.

<u>Response to inadequate core cooling</u>

Inadequate core cooling is caused by a substantial loss of primary coolant resulting in a partially or fully uncovered core. Without adequate heat removal, the core decay energy would cause the fuel temperatures to increase. Severe fuel damage would occur unless core cooling is promptly restored. Reinitiating of high pressure safety injection is the most effective method to recover the core and restore adequate core cooling. If some form of high pressure injection cannot be established or is ineffective in restoring adequate core cooling, then the operators must take actions to reduce the RCS pressure in order for the passive safety injection accumulators and low pressure residual heat removal (RHR) pumps to inject. Analyses have shown that a rapid secondary depressurization is the most effective means for achieving this. If secondary depressurization is not possible, local actions are provided to open SGs PORVs. If primary-to-secondary heat transfer is significantly degraded due to a loss of secondary heat sink, a variety of means are available for establishing secondary heat transfer, including portable equipment and variety of water supplies. If all those actions are not available, or are delayed due to extended time for local actions, then the operators are instructed to start RCPs. The RCPs would provide forced two-phase flow through the core and temporarily improve core cooling until some form of make-up flow to the RCS could be established. However, if the core exit thermocouples (CET) temperatures remain above defined critical temperature, operator would then open primary PORVs and reactor vessel head vent valves to reduce RCS pressure, although inventory losses would be greater. Some injection flow must be established using »feed and bleed« method. If CET temperatures are still greater than defined critical temperature and increasing, and all the actions to cool the core are ineffective, then operators are instructed to enter Severe Accident Management Guidelines.

Response to nuclear power generation (ATWS event)

»Anticipated transients without scram« (ATWS) is an unspecified common-cause failure which preclude control rods from being inserted into the core in response to an anticipated transient which requires a reactor trip. The reactor coolant system conditions at the time the operators identify an ATWS event can be very different depending on the initiating event. Loss of main feedwater, control bank withdrawal at power, loss of AC power, turbine trip, closure of main steam lines isolation valves and a spurious opening of a pressurizer PORV are examples of different ATWS events. The required operators' actions following identification of an ATWS event are the same but the RCS conditions may be very different. Operators must be aware of such system responses and not rely on any signals or indications other than those for reactor trip.

Operators' actions during such an event are to verify automatic actions completed or to perform manual actions for reducing core power. If automatic actions are not effective, any manual reactor trip from MCR is to be actuated, or control rods should be inserted manually. Cutting power supply which keeps control rods electromagnets energized would cause gravity fall of all control rods into the core. Supplemental turbine trip and AFW actuation checks provide consistency with the supporting ATWS analysis. Also borating RCS is an effective way of inserting negative reactivity into the core. Several methods of emergency borating are available from MCR. This action is taken prior to initiating more time-consuming local actions to trip the reactor and/or turbine. If control rods for any reason did not fall into the core, operators would perform RCS heat up, therefore inserting negative reactivity due to negative temperature coefficient, and effectively lowering reactor power by using physical properties of reactor itself, and continuing to borate and adding additional negative reactivity. Possible sources of positive reactivity are also checked and eliminated. Actions include isolation of all dilution paths and identification/isolation of faulted steam generator(s), which may cause an uncontrolled RCS cooldown. Final action is checking on the effectiveness of previous steps by verifying reactor sub-criticality in mitigating the transient prior to exiting the guideline. Until sub-criticality is verified, return to other procedures is not allowed.

At the onset of core damage, the operators would be in one of the symptom based EOPs. With reference to particular tasks, operator actions may be as follows:

1. Restoring feedwater to steam generators (SGs)

In the EOPs, in the event of a loss of feedwater to the SGs, the procedures instruct the operators to establish an alternative source of feedwater. Operators' actions involve manual operation of the auxiliary feedwater pumps or SGs depressurization and the use of feedwater or condensate pumps.

If these actions are not completed, operators perform the actions to initiate bleed and feed and whether it is successful or not they would continue attempting to establish an alternative feedwater source. For example, in the case where the operators fail to establish bleed and feed within the time specified in the success criteria, core damage could occur, but the operators would continue trying to establish an alternative feedwater source. Therefore, the success for feeding the SGs should consider the fact that the action may have been initiated in the EOPs prior to core damage, but not soon enough to prevent core damage, although the consequences would be mitigated. Krško NPP has a variety of alternative methods for injecting into the steam generators if pumps, feed paths, suction sources and control power are not available.

In severe accident management, first priority is to feed SGs, so feeding SGs is a good precautionary measure. Feeding the SG's is addressed throughout the EOPs as a continuous action step (continuous operator task) until appropriate SG level or feedwater flow is restored.

2. Injecting into the RCS

If establishing of a secondary heat sink was not accomplished, operators would establish feed and bleed flow to reactor core. If AC power for emergency core cooling pumps is lost, Krško NPP has also availability to establish injection into RCS using low head portable fire protection pump using suction from variety of borated or un-borated sources (with 3 days fuel supply onsite), although RCS depressurization would be needed.

If adequate core cooling cannot be established or maintained, decay heat is absorbed by the core materials, which eventually melt and relocate downward. The only way to ensure adequate long-term heat removal from the core during a severe accident is to flood the core with water. The injection of water to an overheated core would initially result in the flashing of the water to steam and the removal of heat from the core. If the flow rate of water to the core is large enough, all of the continued decay heat production, along with the excess sensible heat of the core can be removed and the core would eventually be flooded. In the process of providing water to the core, additional oxidation of metals in the core may also occur and the energy released by these reactions must also be removed. Eventually, all of the excess sensible heat would be removed from the core and only the remaining decay heat needs to be continually removed by either continued water addition or by establishing a reflux heat removal process.

3. Injecting into containment

Following the onset of a core damage accident, the containment would contain a substantial amount of water from the lost RCS inventory and possibly from the accumulators if the RCS pressure decreased to allow their discharge. If the accident sequence includes successful usage of the RWST inventory for safety injection and containment spray (if required), then the RWST water would be delivered to the containment, which guarantees spillover to the reactor cavity.

This operator action is only applicable for core damage accident sequences in which the RWST has not been emptied to the containment via either safety injection or containment spray system, although Krško NPP has a variety of options for additional water supply to containment. Several major accidents management benefits can be realized by injecting water into the containment during a severe accident (or as a precautionary measure from severe accident). First, water in the containment sump can be used for ECCS injection or containment spray if it subsequently becomes available. Second, water on the containment floor can quench the core debris following vessel failure and prevent molten core concrete interaction and basement melt-through. Third, fission products released from core debris on the containment floor would be scrubbed. Injecting into containment without establishing long-term heat removal would not prevent containment failure, but according to the analysis would significantly delay containment failure for more than a day.

4. Depressurize the RCS

In the event of a loss of coolant accident in which the RCS pressure remains above the shutoff head of the low pressure SI pumps, the EOPs would instruct the operators to initiate a rapid RCS cooldown and depressurization by dumping steam from the SGs. Further in the accident scenario, the inadequate core cooling procedure instructs the operators to open the pressurizer PORVs to effect a more rapid RCS depressurization. RCS depressurization using pressurizer PORVs may not be available for extended station blackout sequences in which the batteries or air supply are depleted prior to the onset of core damage. Therefore Krško NPP is provided with two onsite portable air compressors which could restore instrument air to PORVs, and portable generators for providing necessary power to motor operated valves (opening letdown path or reactor vessel head valves for depressurization).

However, it could be assumed that core damage has already occurred because the entry condition for the inadequate core cooling procedure is the same as the definition of core damage (i.e., based on core exit thermocouples indication). However, in both cases the operators' actions would be taken and core recovery would occur after core damage by using the low head SI pumps. Therefore, we should consider that RCS depressurization has been initiated in the EOPs prior to core damage, but not soon enough to prevent core damage.

In the severe accident, depressurizing the RCS decreases the potential for a High Pressure Melt Ejection (HPME) and decreases the potential for creep rupture of the steam generator tubes, so operators actions in EOP's are a good precautionary measure if core damage occurs later on. Also, lower pressure would allow more water sources to be injected into the RCS.

5. <u>Late AC power recovery</u>

Recovery of AC power before (during) core damage would permit the operators to reestablish either safety injection or containment cooling. When SBO is first diagnosed, the operators place all equipment in pull-to-lock position. The operators' actions taken after AC power restoration would be energizing the emergency buses, and then placing equipment back in service. Therefore, the procedures provide priority for reestablishing equipment after AC power has been restored. Krško NPP has alternative diesel generators available onsite, which can provide power to a particular AC bus for three days of continuous operation of positive displacement pump, battery charger and associated instrument busses. However, local operators' actions are required starting at the beginning of the SBO.

6. <u>Establishing of containment sump recirculation</u>

The operators' action to establish containment sump recirculation applies when injection into the RCS is performed by using safety injection or when injection into containment is performed by using containment sprays. Without establishing sump recirculation, containment pressure would continue to rise and additional injection would be required when containment is vented or fails. Therefore, it is necessary to establish containment sump recirculation.

6.2.1.2 After entering a severe accident situation following damage to the fuel or even the pressure vessel

6.2.1.2.1 Identification of the risks, cliff edge effects and kinetics of severe accidents

Krško NPP performed sensitivity analysis for different long-term SBO accident sequences, with focus on the containment pressure response following core damage. Accordingly, several scenarios were evaluated, with respect to the RCP seal leakage flow, availability of the AFW TDP and heat sink, possibility of gravity feed from RWST to the containment sump, alternative spray/injection using the alternative sources and containment venting.

Following the loss of all AC power, operators would perform EOP procedure »Loss of all AC power«. RCP seals would lose their cooling and it is assumed that the coolant would be discharged through the seals from the beginning of the transient. The seal leakage rate applied is in accordance with plant data that are using a high temperature o-ring RCP seal packages. If emergency diesel generators fail to start, and auxiliary offsite power sources are not available, operators immediately start establishing an alternative power supply from one of two available mobile diesel generators (located onsite with 3 days fuel supply) for running charging PDP to supply water to RCS and charging one safety batteries to provide instrument and control power. In addition, operators would isolate letdown line by using portable diesel generators for establishing power to motor-operated valves inside containment.

If there are no actions taken for isolating letdown line, it would consequently lead to the opening of letdown relief valve to pressurizer relief tank increasing the coolant loss until RCS pressure decreases below valve set-point. Operators would rapidly depressurize secondary side using SG PORVs (it can also be done by local manual operation), which would lead to primary side cooldown and depressurization. RCS pressure decrease results in injection of accumulators into the RCS until pressure drops below the limit when accumulators should be isolated. AFW TDP does not require electric power and can operate if the SG pressure is appropriate, so it can provide secondary injection for unlimited time. Taking into account that the AF air operated flow control valves are operable (using nitrogen or alternative compressed air supply or manual operation) and that CSTs can be refilled by variety of alternative sources and portable fire pump (including river water), the secondary side heat sink is available during the entire accident. Even if AFW TDP is not operable, Krško NPP has available portable firewater pumps onsite, with variety of injection flow-paths to SGs and variety of water sources. Fire track can also be used. The primary coolant loss is rather low and rapidly decreases when the RCS water level drops below RCP seals leakage elevation (since the break flow is steam). By cooling the RCS, RCP seals are also cooled. Consequently, the core does not uncover within 7 days, so the core, RCS and the containment integrity are not endangered.

First identified cliff edge effect would be **core damage**. No cliff edge effect concerning core damage can be expected if a secondary heat sink is available or RCS feed and bleed method for decay heat removal and lost inventory replacement is established.

Second identified cliff edge effect would be **reactor vessel failure**.

If core damage has occurred and there is no water injection into the reactor vessel, continued process of core degradation and melting would result at some point in a relocation of melted material to the lower plenum of the reactor vessel. The response of the reactor vessel wall to the imposed stress would be possible melting of the inner surface, resulting in reactor vessel failure. It is expected that system pressure boundary will fail before vessel failure due to natural circulation of hot gases inside the RCS piping. Analyses for NPP Krško confirm that RCS would be depressurized and if the reactor vessel afterwards fails, the challenges to the containment are minimized. Also NPP Krško wet cavity design will successfully mitigate the consequences of reactor vessel failure. See also section 6.2.2.4.2.

Containment failure is also identified as cliff edge effect.

Opening of containment isolation valves on sump recirculation lines by using onsite available portable diesel generators and gravity flood the containment from RWST, the time of containment failure can be significantly delayed, thus extending the time available for other mitigating solutions, even though no containment spraying is established.

However, the most effective means to protect containment integrity during SBO and loss of ultimate heat sink is spraying the containment atmosphere with alternative portable fire protection pumps (or fire track), which can also effectively use RCP fire protection spray nozzles beside normal containment spray lines. By doing this, containment will not fail within 7 days.

Another option, if no containment spray is available, is to perform containment venting prior to containment failure. Guidelines direct plant evaluation team staff for containment venting, using various flow-paths, because preserving the containment integrity is of the highest priority for mitigating severe accident in a long term. Containment venting would reduce the pressure in the containment at the cost of releasing fission products from the containment atmosphere in a controllable way. Note that the objective is only to reduce containment pressure far enough to mitigate a severe challenge, not to depressurize the containment. Because of the fission product release, venting is considered the last option to prevent containment failure. Detailed description of cliff edge effect concerning containment failure is given in sections 6.2.2.1.2, 6.2.2.2.2 and 6.2.2.5.2.

Conclusions are as follows:

- Providing whichever sources of heat sink (auxiliary feedwater, normal feedwater, condensate pumps, service water pumps, fire protection pumps or alternative portable fire protection pumps or fire track) and assuring its long-term availability (condensate water, service water or variety of alternative sources) can successfully prevent core damage for extended period of time (> 7 days) and therefore stop the progression of containment accident.

- If secondary heat sink is not available, replacing of RCS inventory loss by feed and bleed method can also successfully prevent core damage for extended period of time (> 7 days) and therefore stop the progression of containment failure accident.
- Injecting into the reactor vessel, even with damaged core, will successfully prevent or delay reactor vessel failure.
- Spraying of water into the containment atmosphere through various compartments in the containment can reduce the pressure rise even if this action is performed late in the accident with reactor pressure vessel failed. Spraying the containment atmosphere can successfully prevent containment failure for extended period of time (> 7 days).
- Possible means to inject water to flood the containment (RWST gravity draining or alternative injection methods) can prolong time to the containment failure even if ultimate heat sink is lost, and substantially decrease the amount of hydrogen produced by molten core-concrete interaction (MCCI) in the containment (app. 2 days).
- Venting of the containment has a positive effect in preserving the containment integrity and decreasing the content of hydrogen in the containment (beside the negative impact, which is radioactivity release to the environment).

Risks identified in a severe accident, which could apply to Krško NPP are:

- Primary inventory loss
- Core uncovery and heatup
- Core melt progression
- Hydrogen production during in-vessel core degradation
- Natural circulation and heatup of reactor system structures
- Reactor system piping failure
- Core melt interaction with a vessel wall
- Direct containment heating
- Vessel thrust at vessel failure
- Debris coolability and molten core concrete interaction
- Hydrogen behavior in containment
- Steam explosions
- Loss of spent fuel pool inventory

A brief description of risks and their applicability to Krško NPP are presented below.

Primary inventory loss

Conditions leading to overheating of the core materials result from a loss of primary coolant. Such a loss can result from either break in the primary system or a loss of the heat sink for the primary system causing inventory loss via safety/relief valves. The primary inventory loss stage of accident can vary in duration from few seconds (e.g. large reactor system break with ECCS failure) to several hours (e.g. a secondary side initiated transient, small breaks in the primary system or RCP seal degradation).

Core uncovery and heatup

If the inventory loss is sustained, core uncovery eventually occurs, and heatup of the uncovered portion of the core fuel rods begins. This overheating results in oxidation of the cladding material and eventually loss of geometric integrity of the core. This process in general had been termed »core degradation«. During the heatup phase, the core geometry is basically unchanged.

Core melt progression

As core temperatures continue to rise, substantional fuel rod damage occurs, including mechanical failure of the material. Once melting begins the core geometry starts to change significantly. After control rods (Ag/In/Cd), stainless steel and zircaloy melts and relocates at the lower cooler parts of the core where it may refreeze, and cause partial or complete blockages. Thus the early melt progression phase involves mainly metallic material relocating downwards where it refreezes. As temperature continues to rise, a fraction of the UO_2 dissolving in the liquid Zr becomes significant, and adds to the solidified metallic mass. It is expected that the free standing stacks of fuel pellets basically in original geometry with no structural strength would collapse and fall into the solidified molten metal bed, leaving a void in upper core regions. Continued heatup of the solidified mass would occur due to fission product decay heat, and remelting would start in the center of the bed, with solid crust surrounding the structure. Continued heatup and melting would finally weaken and fail the surrounding crust, leading to relocation of the molten mass to a lower plenum of the vessel.

Hydrogen production during in-vessel core degradation

In a severe accident involving significant core degradation, hydrogen is produced primarily by the oxidation of zirconium in the fuel cladding. During fuel heatup, melting and relocation phases, a significant fraction of the zirconium in the core may be oxidized. The oxidation reaction is strongly exothermic, releasing a heat input which can be several times greater than decay heat. Analyses predict that reaction would rapidly progress, although zirconium – water reactions do experience some limiting mechanism due to unavailability of additional steam to continue the reaction, and the covering of the unreacted zirconium which forms the lower portion of the fuel rods by molten core debris as it relocates downwards from above. The presence of hydrogen in the reactor coolant system is of limited importance. However, once released to the oxygen rich containment atmosphere, the released hydrogen represents a potential challenge to containment integrity.

Natural circulation and heatup of reactor system structures

Following core uncovery in the progression of a severe accident sequence, natural circulation of superheated steam and hydrogen can occur in the reactor vessel and reactor coolant system, because of small differences of gas densities between various regions as a result of different heat losses to the structures in each region. Natural circulation of gases is important because it transports heat and slows heatup rate of the core, and it causes the heatup of reactor system structures which can cause failure of reactor system pressure boundary before vessel failure. Analyses for Krško NPP confirm that RCS would be depressurized and if the reactor vessel afterwards fails, the challenges to the containment are minimized.

Reactor system piping failure

If a severe accident is not terminated by restoring cooling to the core, a failure of the reactor system pressure boundary could occur. Creep rupture is rupture of RCS structure occurring due to heatup caused by natural circulation flow. Potential for the reactor system creep rupture is highly dependent on the amount of in-vessel hydrogen generation and the time interval after core relocation. Mechanisms which delay vessel failure result in higher temperatures for RCS piping. The analyses also show that creep rupture failure of the RCS piping prior to reactor vessel failure have positive impact on containment and radiological challenges, because if reactor vessel fails at high pressure, a molten core debris can disperse which could inflict direct containment air heating and large fission product releases.

Analyses for Krško NPP show that RCS hot leg failure is likely to occur for most severe accident sequences occurring at high pressure, if the system is not manually depressurized first. Creep rupture of Krško NPP SG tubes is not likely to occur since temperatures in tubes do not approach failure values in any analyses. Anyway, due to possible breach of containment barrier with U-tube creep rupture, a NPP strategy in mitigating the severe accident gives the highest priority to fill the steam generators with water, and to depressurize the RCS (conditions for U-tube rupture are significant higher pressure in the RCS and dry steam generators).

Core melt interaction with a vessel wall

The continued process of core degradation and melting would result at some point in a relocation of melted material, at high temperature, to the lower plenum of the reactor vessel. The relocation would lead to stress in the vessel head wall. The response of the wall to the imposed stress would be plastic and creep strain, and possible melting of the inner surface.

Failure of the pressure vessel lower head following relocation of molten debris can occur by a number of mechanisms, including:

- Melting of welds attaching in-core instrumentation tubes and subsequent ejection of the tube(s);
- Jet attack of the vessel wall leading to localized failure;
- Delayed creep failure of the vessel wall following formation of a convective debris pool.

Direct containment heating

For sequences in which the reactor vessel fails at high pressure, the possibility exists that a significant portion of core debris may be finely fragmented and dispersed out of the cavity into the upper volume of the containment where it may rapidly transfer thermal and chemical energy to the containment atmosphere and threaten the containment integrity, leading to over pressurization.

A detailed evaluation of the potential for direct containment heating in the Krško NPP has been performed. Wet cavity design of Krško NPP would significantly mitigate the consequences of direct containment heating.

Vessel thrust at vessel failure

This phenomenon is associated with the behavior of reactor vessel immediately following high pressure melt injection.

Krško NPP analyses, assessing a possibility that the vessel itself becomes a missile, which could threaten containment integrity due to large force acting during the blowdown phase, have confirmed that there is a considerable margin between the upward force experienced by the vessel and its available restraining force. For all failure modes analyzed, reactor vessel motion poses no threat to containment integrity.

Debris coolability and molten core-concrete interaction

When vessel failure occurs, the molten core material would be released from the vessel and would relocate to the lower containment region. Subsequent behavior in the containment depends strongly whether the ex-vessel debris bed is quenched and cooled. The term quenched refers to removal of sensible and latent heat, together with any oxidation reaction from the debris in short-term following its ejection. The term cooled refers to long-term removal of decay heat. If the debris is not quenched and cooled, the molten core-concrete interaction (MCCI) is expected to occur, resulting in non-condensable gases which pressurize the containment, the combustible gases would be released, fission product aerosols would be released and erosion of basement could occur, potentially leading to long-term containment failure.

Krško NPP with wet containment design and early containment flooding strategies can successfully mitigate that challenge. If heat removal is not provided in long-term, it would cause containment pressurization. No further hydrogen generation and little or no fission product releases from debris would occur. Containment pressure versus time during the severe accidents of core meltdown, reactor pressure vessel failure and MCCI in the reactor cavity was investigated. The accidents are initiated respectively by a large break LOCA, a small break LOCA, and a small break LOCA on the RCP seal. The interaction between the molten core and concrete generates large quantity of non-condensable gases, which causes a significant pressure increase. The simulation showed that the atmosphere pressure in the containment does not reach the design pressure during the first 72 hours following the reactor pressure vessel failure.

Hydrogen behavior in containment

Hydrogen produced in-vessel would normally be released through break location of reactor system, vessel failure point or via the pressurizer relief system (through rupture disc from PRT). The behavior of hydrogen in containment is influenced by numerous factors, which may affect both distribution of the gas and the likelihood of the consequences of combustion. The degree to which hydrogen is mixed with other gases is important, because relatively good mixing minimizes the danger of localized buildup of potentially high concentrations, and global deflagrations are likely if flammability limits are reached. Deflagration is a combustion process in which the combustion front (flame) moves at subsonic velocity with respect to unburned gas. Pressure may increase sharply in time, but it is uniform through the volume which implies that loading on structures are considered to be static loadings since the pressure loading is over a period of several seconds. If containment sprays and/or fan coolers are initiated, mixing becomes even stronger, due to turbulence and convective flows set up by spray and fans, but there are also other mechanisms which encourage gas mixing during an accident.

Detonation is another type of combustion which may occur, in which flame propagate at supersonic velocity. In this case it will be a strong spatial variation in pressure due to production of shock valves. Loading to structure would consider being impulse loadings.

The type of combustion (deflagration or detonation), and the extent of combustion (the amount of fuel consumed), depends upon the concentration of hydrogen, oxygen and inert gas (steam, nitrogen), the initial temperature and pressure, geometry, presence of ignition sources and turbulence.

Detonations, including transitions from deflagration to detonation (DDT) are unlikely to occur in Krško NPP containment due to the limited hydrogen concentrations reached. In addition, Krško NPP has a large containment, and is equipped with redundant spray and fans system which maximizes containment gas mixing. Also there is a possibility for using portable firewater pumps with variety of suction sources discharging via spray system connections to containment. RCP fire protection system spray nozzles are also used as additional option for spraying the containment. The TSC uses hydrogen concentration measurement instruments, and specifically developed curves of hydrogen behavior for mitigating a severe accident. An evaluation of Krško NPP potential for a severe accident hydrogen detonation concludes that detonation of gas mixtures in containment is not expected.

Steam explosions

A vapor explosion is a process in which a vapor production occurs at a rate larger than that in which surrounding media can acoustically relieve the resulting pressure increase, resulting in the formation of a shock wave. Steam explosion within the primary system has been considered, potentially damaging both the primary and containment building. It is postulated to occur following relocation of molten core debris into the water filled lower plenum of the reactor vessel. In this scenario, in-vessel steam explosion ruptures reactor vessel head which becomes missile with sufficient velocity to fail containment.

Reactor vessel at Krško NPP is protected by concrete missile blocks, and probability of such an explosion is so low that it is not considered to be likely.

Another phenomenon is ex vessel steam explosion which may occur in the progression of a severe accident should debris be discharged from the vessel into a pool of water. This could cause a rapid pressurization of the containment.

Based on reviews and evaluations, conclusions are drawn regarding the importance of steam explosion for Krško NPP: an in-vessel steam explosion which leads to reactor vessel-containment failure is not possible; ex-vessel steam explosions would cause no additional changes to containment integrity; containment pressurization due to the steam explosions has no impact and ex-vessel shock waves pose no threat to containment integrity.

Loss of spent fuel pool (SFP) inventory

If SFP cooling is lost, water in the pool would heat to boiling point and start to evaporate. With time, water level would drop below the top of the stored fuel elements. Cooling of fuel assemblies can be possible by using portable sprays. If sprays are ineffective, overheating of fuel assemblies would be expected when water level drops to 20% of fuel assembly high within racks. From then on fuel degradation is expected yielding volatile gas releases, possible exothermic zirconium oxidation and hydrogen generation.

Time to reach the limit depends on total heat power of the fuel elements stored in the pool. If bounding case is considered taking into account maximum decay heat load with core unloaded from the reactor immediately after 18-months fuel cycle it would take approximately 6 days to that point.

6.2.1.2.2 Identifications of existing measures for different risks

The severe accident management guidelines (SAMG) provide guidance for managing the in-plant aspects of an accident that progresses to core damage and in which the design bases of the plant are grossly violated. Because EOPs prioritize core damage prevention, they may not be appropriate after core damage. Thus, the EOPs are terminated when the transition from EOP to SAMG is made. Refer to Figure 17.

EOPs are rule-based procedures. A rule-based procedure requires that a very specific action would be taken for a given plant condition. It requires little or no evaluation of unintended consequences or negative impacts that might arise from taking the action. Most of the SAMG could not be developed as rule-based procedures. The SAMG include variables in the applicability and magnitude of positive and negative impacts associated with a given action. The SAMG require evaluation and decision-making processes to select proper actions for implementation. Therefore most of the SAMG consist of knowledge-based guidelines. A knowledge-based guideline does not mandate that a particular action is taken for a given condition. Rather, it identifies potential strategies for a given condition and leaves it up to the user to decide the best course of action under the circumstances. The consensus within the SAMG developer was that the plant evaluation team is best-suited for using knowledge-based guidance, which requires an essentially engineering approach.

In the EOPs, the emphasis is on preventing core damage. In the SAMG, the presumption is that core damage has already occurred. Therefore, when the transition from the EOPs to the SAMG is made, priorities shift – from preventing core damage to preserving the containment fission product barrier and arresting the progression of core damage.

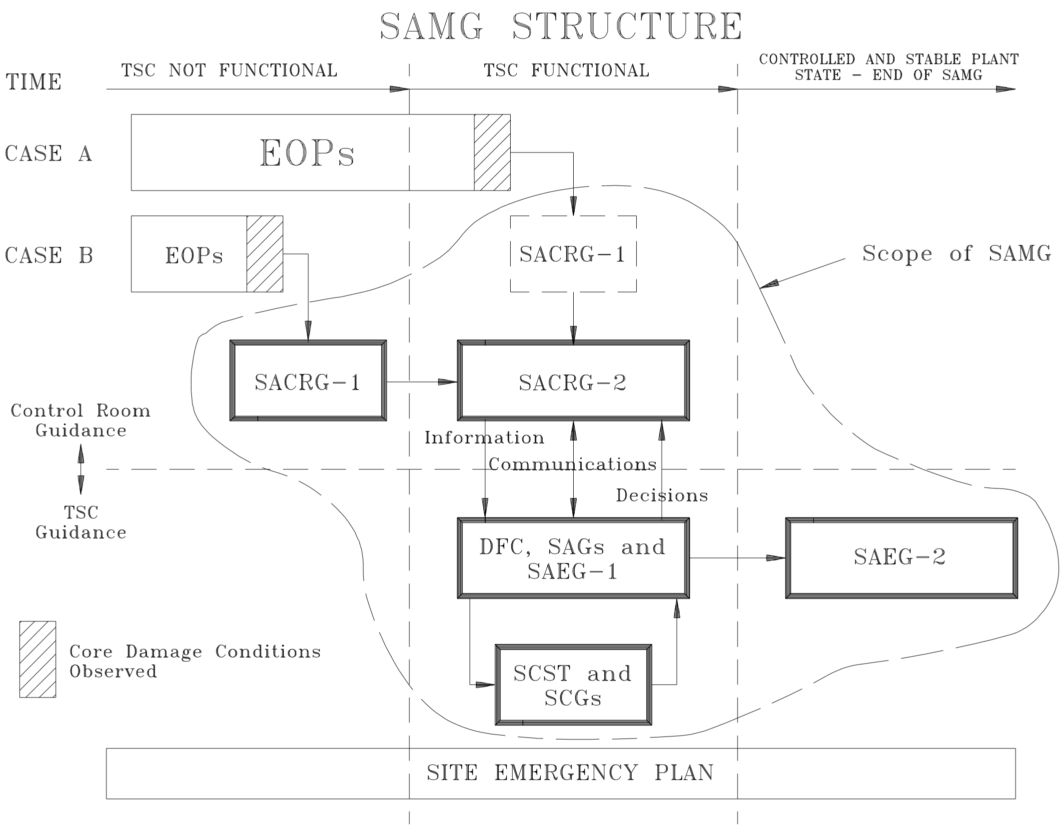


Figure 17: Graphic presentation of transition from EOPs to SAMGs (before and after TSC is functional) and SAMG structure

In the event of core damage accident there are three major types of response actions:

- Control and termination of fission product releases
- Prevention of severe challenges to the containment fission product boundaries
- Recover core cooling

Secondary actions are to:

- Minimize fission product releases while achieving primary goals;
- Maximize equipment and monitoring capabilities while achieving the primary goals.

The Westinghouse Owners Group (WOG) SAMGs were developed to enhance the capability of the plant emergency response staff to:

- Diagnose the plant status during an accident which progresses to core damage,
- Perform a systematic and logical evaluation of possible severe accident strategies to choose the optimal strategy at any point in a severe accident and
- Evaluate the effectiveness of a severe accident strategy once it is implemented.

SAMG are primarily used for evaluators, not for implementers (operators). An evaluator is a member of the Plant Evaluation Team (PET) tasked with any of the following activities:

1. Diagnosing conditions that require entry into specific guidelines;
2. Assessing availability of equipment to perform the required strategy;
3. Evaluating the positive and negative impacts of strategies presented in certain guidelines;
4. Providing the recommended actions;
5. Interpreting the response of plant parameters following strategy implementation;

6. Assessing the effectiveness of implemented strategies and determining whether additional mitigation is needed;

SACRG-1

SACRG-1, »SAMG Control Room Guide – Initial Response« is the initial SAMG used by the control room staff. The control room staff enters SACRG-1 from the EOPs when conditions indicate that significant core damage is occurring. The control room staff is using SACRG-1 until the TSC is operational and the PET is ready to use SAMG.

Analyses show that for all accident sequences leading to core damage, the control room staff would be attempting to implement EOP procedures at the time of core damage. More specifically, the control room staff will be either in Loss of core cooling procedure (for all accident sequences except a Loss of all AC power) or Loss of all AC power procedure (for SBO accident sequences). Although there is a transition from Response to nuclear power generation ATWS procedure to the SAMG, it is expected that the control room would be using Loss of core cooling procedure at the time of damage from ATWS event (refer to section 6.2.1.1.1.). Since the operators close the EOPs when entering SAMG, SACRG-1 includes many of the steps that are in the EOPs for Response to inadequate core cooling. Additional steps, not related to core cooling, are included to provide a broader focus to protection of the fission product barriers.

Actions in SACRG-1 are:

1. Taking manual control of equipment to prevent automatic actuation of inactive equipment
2. Controlling hydrogen equipment – the hydrogen measurement system is put into operation, and hydrogen recombiners are switched off to prevent ignition of hydrogen if it escapes to containment. The guidance can be performed in a relatively short period.
3. Providing sufficient containment water inventory to allow ECCS recirculation capabilities and to mitigate possible consequences of vessel failure which could impact containment integrity in relatively short period (less than a day). Analyses showed that flooding containment in an early period of a severe accident significantly improves containment integrity to more than a day (if no other action is made with progressing accident). It is also possible to flood the containment without AC power available, using gravity line from RWST to containment sump, by using small portable generators stored on-site for providing necessary power to motor-operated valves inside containment, and by manual operation of valves outside containment.
4. Controlling (depressurizing) RCS pressure to prevent high pressure vessel failure or steam generator tube failure by opening pressurizer PORVs. PORVs are powered by instrument air. Two portable onsite stored air compressors can be used for alternative instrument air supply, with fuel supply for 3 days operations.
5. Continuing attempts to restore core cooling. Following RCS depressurization, additional methods for injecting water may become available. If SBO is in progress, it is possible to use positive displacement pump which can be powered from one of two alternative mobile diesel generators stored on-site with fuel supply for three days of operation (attempts for establishing alternative power start in Loss of all AC EOP), or by using other alternative filling methods as described in SAG-3, i.e. a portable fire protection pumps or fire truck with variety of borated or un-borated water sources with discharge connection to safety injection line.
6. Controlling containment pressure to avoid hydrogen severe challenge conditions and addressing possible ignition sources. Containment should be maintained enough steam inert, but not over-pressurized.
7. Controlling steam generator water inventory; these actions are already implemented in EOPs.
8. Controlling and establishing effective containment and secondary system pressure boundaries.

The last step of SACRG-1 returns the Control Room to the point where the status of the TSC is checked again.

SACRG-2

A second control room guideline is used when the TSC is staffed and functional, and monitoring the plant status. It provides actions to respond to a severe accident in which the core may be damaged.

Diagnostic Flowchart (DFC)

On entry to the SAMG, the TSC begins immediately to monitor the DFC.

If a setpoint is exceeded in DFC, the TSC implements the corresponding Severe Accident Guideline (SAG), taking into account orders of priority.

A list of all Krško NPP SAGs is in Table 2. Prioritization of SAMG strategies is shown on Figure 18.

Table 2: List of Krško NPP TSC Severe Accident Guidelines (SAG)

TSC guidelines and associated DFC parameter		
Guideline	**Description**	**DFC parameter**
SAG-1	Inject into steam generators	Water level in all SGs
SAG-2	Depressurize RCS	RCS pressure
SAG-3	Inject into RCS	Core temperature
SAG-4	Inject into containment	Containment water level
SAG-5	Reduce fission product releases	Site releases and SFP temperature
SAG-6	Control containment conditions	Containment pressure
SAG-7	Reduce containment hydrogen	Containment hydrogen concentration
SAG-8	Flood containment	Containment water level

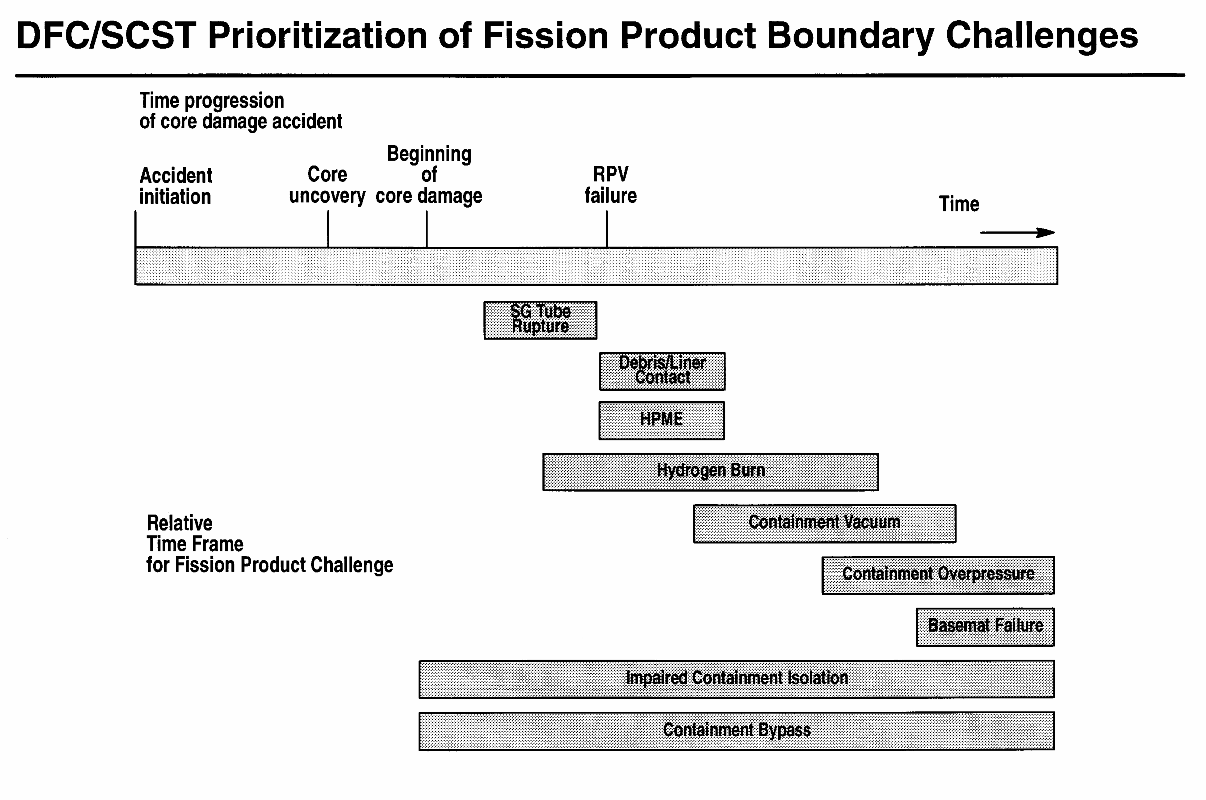


Figure 18: Diagnostic prioritization of SAMG strategies based on prioritization of fission product boundary challenges

Inject into steam generators (SAG-1)

The purposes of injecting into the SGs are to protect the SG tubes from creep rupture (preventing breaching containment fission product barrier), to scrub fission products that enter the SGs via tube leakage and to provide a heat sink for the RCS.

There are many pumps that can inject water into the SGs, including two AC motor driven and one steam driven auxiliary feed pumps, main feed pumps, condensate pumps, service water pumps, AC firewater pump and one diesel driven firewater pump (from plant fire protection system having suction from river), onsite mobile firewater pumps and submersible firewater pumps for the Sava river suction (which are equipped with a three-day fuel for continuous operation). Normal suction sources from condensate storage tanks and service water are extended with additional sources as fire protection tank, water treatment tanks, pretreated water tanks, condenser hotwell, city water, circulating water tunnel, and the Sava river water. Also, additional feed paths are introduced as auxiliary feedwater piping, two connections at main feedwater piping, fire protection piping, water treatment piping, blowdown piping and additional city water piping. Most of the flow-paths are equipped with quick connections for fire protection hoses which give a variety of flow-path choices. A list of criteria necessary to operate the SG feed pumps has been developed. Thus, the TSC can determine which pumps, flow-paths and water sources are available to inject into the SGs and what are the equipment limitations which could prevent pumps from injecting into the SGs. Some of the pumps may be prevented from injecting into the SGs due to high SG pressure. If this is the case, then equipment that may be used to depressurize the SGs must be evaluated. SGs have to be depressurized by SG PORVs or using steam dump system to allow low pressure feed injection. PORVs can be operated also by local manual action.

Depressurize the RCS (SAG-2)

Lowering of RCS pressure during a severe accident is one of the top priorities of severe accident management to prevent a high pressure melt ejection, to prevent creep rupture of the SG tubes when the SGs are dry, to allow RCS makeup from low pressure injection sources and to maximize RCS makeup from any injection source.

The purpose of this action is to determine if any means for depressurizing the RCS are available. There are several ways of depressurizing the RCS, including pressurizer relief valves, pressurizer spray, SG depressurization, and other RCS vent paths. Most of the valves require some type of motive force to operate (either AC power or instrument air) and control power (DC power), which may not be immediately available during a severe accident. Since the operators are instructed in EOP procedures to depressurize the RCS and SGs once core exit temperatures exceed 650 °C, either an operator error, equipment failure or transition from ECA-0.0 must have occurred if the RCS is still pressurized. Design of NPP allows direct and indirect depressurization of RCS. Direct RCS depressurization could be achieved by pressurizer PORVs (which requires instrument air), normal pressurizer spray, auxiliary pressurizer spray, RCS head vent, letdown and excess letdown flow-path. Indirect depressurization covers the usage of SG PORVs or steam dump. Krško NPP has 2 additional mobile air compressors stored on-site, which can be used as additional instrument air source for direct RCS depressurization with pressurizer PORVs.

Inject into the RCS (SAG-3)

The purposes of injection into the RCS are: to remove stored energy from the core when it has been uncovered, to provide an ongoing decay heat removal mechanism, to prevent or delay vessel failure and to provide a water cover to scrub fission products released from the core debris.

There are at least three types of pumps that can be used to inject into the RCS during a severe accident; charging pumps, high head safety injection SI pumps, and low head (residual heat removal) SI pumps with suction from borated RWST and containment sump, boric acid tanks, reactor makeup water storage tanks, boron recycle holdup tanks, spent fuel pool and reactor makeup water storage tank. To get to severe accident conditions, the ability to use these pumps was either lost or severely degraded. Losing all pumps would require a common cause failure such as SBO or loss of component cooling water. Degraded flow would occur if the RCS could not be depressurized and only a small amount of charging flow entered the RCS. In either case, some actions must be performed before adequate injection is available to quench the core. Alternative injection pumps and methods that can be used to inject into the RCS during severe accident are: boric acid transfer pumps, reactor makeup water pumps and have suction capability either from boric acid tank, reactor makeup water storage tanks, boron recycle holdup tanks, spent fuel pool and reactor makeup water storage tank. Portable fire protection pumps and fire truck pumps stored on-site with fuel supply for at least three days are alternative means for injecting into RCS. Injection into the RCS with portable fire protection pumps is provided with spool piece on the suction side of safety injection pump, therefore providing large varieties of flowpaths to RCS (hot leg, cold leg and reactor vessel). Suction sources to portable pumps could be used from almost all plant available water sources (identical to SAG-1), including boric acid tanks. Core recriticality with unborated water is not an issue with lost core geometry.

Inject water into the containment (SAG-4)

The purposes of injection into the containment are: to prevent or mitigate the consequences associated with MCCI, to scrub fission products released from ex-vessel core debris and to allow ECCS recirculation.

Possible means of injecting water into the containment are: containment spray pumps, gravity feed from RWST to containment recirculation sump, ECCS and reactor coolant pressure boundary break as it is addressed in SAG-3 (Inject into the RCS) and portable fire protection pumps.

Injection flow-paths are through containment spray header using containment spray pumps or portable fire protection pumps, gravity feed through recirculation spray or ECCS lines to the containment sump (it is possible to provide power to the sump isolation valves by portable diesel generators), through fire protection lines for spraying reactor coolant pump and through ECCS. Portable fire protection pumps are stored onsite with fuel supply available for three days of operation. Fire protection track is available onsite as well. There is a variety of suction sources for injecting water into containment, as used in SAG-1 »Inject into the steam generators«.

Reduce fission product releases (SAG-5)

The purpose of reducing fission product releases is to protect the health and safety of the public. Following the onset of core damage, fission products will be released from the cladding gap and possibly from the fuel matrix, and the fission products will be released either to the containment (through a break in the RCS or pressurizer relief or safety valves), to the SGs (through a tube leak or rupture), to the auxiliary building (through ECCS break located outside containment), or to the containment annulus.

Another concern is a loss of SFP cooling. During prolonged loss of SFP cooling a loss of SFP inventory and fuel uncovery is possible.

Fission products in the containment can be released to the containment annulus or to the atmosphere via unisolated containment vent or purge valves or from a hole in the containment. Fission products in the SGs can be released to the atmosphere via unisolated SG PORVs, safety valves, from the condenser air ejector system, or from a leak in the steam supply system. Fission products in the auxiliary building can be released to the atmosphere via the auxiliary building ventilation system. SAG-5 was created to address the mitigation of fission product releases during the severe accident. Reducing containment releases can be done by operation of containment spray and containment fan coolers, and isolation of releasing flow-path.

Using the containment spray (CI) pumps or the reactor containment recirculation fan coolers (RCFC) can reduce fission product releases due to fission product scrubbing and containment pressure reduction. Suction sources for CI system are RWST and containment sump. Available makeup water for these sources is identified in SAG-4. Both systems require AC power to operate, cooling water for CI pumps sealing, component cooling water for RCFCs. If AC power is not available, portable fire protection pumps can be used for CI, using a variety of suction sources.

Using the containment annulus negative pressure fans can reduce fission product release in the case that containment leaks to containment annulus (intermediate space between steel containment and concrete reactor building outside structure). They are effective as long as filters are active.

Dumping steam from a ruptured SG to the condenser can be an applicable means of reducing fission product releases to the atmosphere in Krško NPP. If the ruptured SG is filled, the water over the break location should be quite effective in scrubbing volatile fission products, and the only remaining fission products being released in significant quantities would be noble gases. Since noble gases could not be scrubbed in the condenser, establishing condenser steam dump would not reduce fission product releases.

If the ECCS pumps or the CI pumps are in recirculation mode (e.g., using water from containment sump as a source) it can result in the release of fission products to the auxiliary building. The releases should not exceed the site emergency level limit due to normal leakage from the ECCS and CI system. However, the releases may exceed this level if there is a break in the recirculation lines. If this is the case, then the TSC will need to determine if the release can be mitigated without stopping ECCS or CI spray operation in recirculation mode.

To prevent challenges regarding SFP cooling, temperature of the SFP is monitored in this guideline. High temperature (close to boiling) of SFP water will direct TSC to use mitigating actions, consisting basically of different methods of adding water to SFP. Water is normally added to the SFP from RWST or reactor makeup water storage tank. Provision is also made on the SFP skimmer system with universal firefighting connection. Through this connection portable fire protection pumps can pump water from virtually any available water source onsite, as well as from the Sava river. In addition to that, water can be sprayed over the pool, using the same pumps and firefighting nozzles, also located onsite. Diversity of these methods yields high level of confidence that level in the SFP would not drop below the top of the fuel elements and allow cladding temperatures to rise to the level where hydrogen would generate.

Applicable system to mitigate possible fission product releases from auxiliary building is auxiliary building charcoal cleaning exhaust system. This system is interconnected with the SFP charcoal cleaning exhaust system. Interconnection allows airflow rates in excess of these systems filter plenum capacity to be diverted to another system.

The function of SFP charcoal cleaning exhaust system is to mitigate possible SFP radiological releases. However the efficiency will be reduced due to the additional moisture.

Control containment conditions (SAG-6)

The purposes of controlling containment conditions are to prevent a challenge to containment integrity due to high containment pressure, to prevent a challenge to containment penetration seals due to high containment temperature, to minimize the challenge on containment equipment and instrumentation due to a harsh containment environment, to reduce the airborne fission product concentrations, and to mitigate fission product leakage from containment. There are two generic containment heat sinks identified that are capable of depressurizing the containment to near ambient conditions following a severe accident: the containment spray system and containment recirculation fan cooler units. Both systems are safety-grade and would normally be available during a design basis accident. However, the plant condition that caused a severe accident may also impact the availability of the spray and fan coolers. Therefore, this guideline reviews the criteria necessary to operate the spray and fan coolers to determine if a containment heat sink can be established.

Krško NPP has alternative option if AC power is not available for CI and RCFCs, by the use of portable fire protection pumps of fire truck. Also beside RWST and containment sump used as a normal containment spray suction source there is a variety of water sources onsite: fire truck tank, water treatment tanks, fire protection tanks, condensate storage tanks, pretreated water tanks, condenser hotwell, city water, circulating water tunnel, and the Sava river water. Additional flow-paths are also provided: fire protections hoses connections to containment spray discharge lines and to RCP fire protection spray nozzles.

Reduce containment hydrogen (SAG-7)

Following core uncovery, the core would heat up and the fuel cladding would oxidize in the presence of steam. One of the products of the cladding oxidation reaction is hydrogen, which can accumulate in the RCS or in the containment if a venting pathway exists from the RCS. Following significant core damage, it is very likely that hydrogen concentration would reach 4% to 6% in the containment. Consequentially, the hydrogen can ignite and cause a spike in the containment pressure and temperature. The purpose of reducing containment hydrogen is to prevent hydrogen from accumulating to the point where the containment may be severely challenged. This is achieved by using one of two methods: intentionally igniting the hydrogen in containment, and using hydrogen recombiners.

The fastest way of reducing the containment hydrogen concentration is to intentionally ignite the hydrogen. This will cause a hydrogen burn that spikes containment pressure and temperature. However, once the pressure and temperature spikes are over, then there would be no more short-term challenges to containment integrity due to a hydrogen burn unless additional zirconium-water reaction occurs. The burn is not expected to jeopardize equipment and instrumentation in the containment based on large-scale tests. Sparks caused by following equipment can initiate burning intentionally: 1) reactor building recirculation fan coolers; 2) reactor building charcoal cleanup fans; 3) electric hydrogen recombiners when hydrogen concentration in containment atmosphere reaches more than 6% volume (because of overheating due to the exothermic hydrogen-oxygen reaction).

Containment hydrogen concentration of 4% represents a design limit for recombiners; but it does not represent a real limit for recombiners overheating. In the setpoint calculation the upper limit of operability of the hydrogen recombiners is 6%. During sever accidents the increase of hydrogen above 4% volume limit concentration can occur and based on decision process to identity the appropriate strategy, electric hydrogen recombiner can be used: to continue removing hydrogen from containment atmosphere as it is normally suggested or as ignition source to intentionally burn the hydrogen in containment if flammability limits have been reached. It is noted that the entry condition to SAG-7 is chosen such that the gas mixture may be flammable, but in case of ignition, there would be no threat to containment integrity. Under these conditions, use of recombiners, either for recombination, or for providing an ignition source, is appropriate. Containment conditions where flammable mixtures exist, and where combustion could threaten containment integrity are covered in severe challenge guideline (SCG) 3. In this case, ignition sources, including electric recombiners, would be isolated.

Since Krško NPP median containment failure pressure is at 0.79 MPa, with the peak pressure line at maximum 0.68 MPa, it can be estimated that the containment failure is not expected to occur due to hydrogen burn, even with very high zirconium oxidation fraction (i.e., if 100% zirconium oxidizes, 530 kg of hydrogen is produced).

According to the calculation, hydrogen in the containment will burn if the following two conditions are true: the hydrogen concentration is above 4 volume percent, and the steam concentration is below 55 volume percent. Based on these facts a strategy of preventing hydrogen from igniting can be used for maintaining the containment steam inert by any of the two methods: intentional steaming of the containment or stopping the active containment heat sinks. Another action that can be taken to prevent a hydrogen burn is to isolate all potential ignition sources in the containment.

Flood containment (SAG-8)

The purpose of flooding the containment is to establish cooling of the core material as a long-term strategy when other strategies have been ineffective. Specifically, if the vessel has failed, then the containment may need to be flooded to a level that ensures all core material remaining inside the vessel is covered with water. Three major benefits can be released by flooding containment to submerge the cover material remaining in the reactor vessel: any core material which remains in the vessel after reactor vessel failure would be cooled; water on the containment floor may quench the core debris following vessel failure preventing basement melt-through; fission products released from core debris on the containment floor would be scrubbed.

The amount of water necessary to flood the containment to the elevation of the top of active fuel, which would ensure in-vessel core debris cooling, is approximately 3 RWST volumes. The major impact that a flooded containment can have on the accident progression is described in severe accident guideline SAG-4.

Possible ways of injecting the water into the containment are: two containment spray pumps, gravity feed from RWST to the containment recirculation sump (it is also possible to locally energize sump line isolation valves by portable diesel generators stored onsite), alternative flow-paths through ECCS and RCS pressure boundary breaks as it is addressed in SAG-3, and portable fire protection pumps. Injection flow-paths are through the CI header (also available by portable fire protection pumps), through discharge of ECCS pumps to the RCS, through CI and ECCS recirculation lines to the containment sump, and through fire protection lines for spraying RCPs. Onsite stored portable fire protection pumps and fire protection truck are available, with fuel supply for three days of operation. Beside borated water sources (i.e., RWST, boric acid tanks, borated water hold-up tanks) and reactor make-up water storage tank, there is a variety of un-borated water sources available onsite: fire track tank, water treatment tanks, fire protection tank, condensate storage tanks, pretreated water tanks, condenser hotwell, city water, circulating water tunnel, and the Sava river water.

Severe Challenge Guidelines (SCGs)

If SCG setpoint is exceeded in Severe Challenge Status Tree (SCST), the TSC stops monitoring the DFC (i.e., evaluating SAG) and refers to appropriate SCG. The difference between SCGs and SAGs is that immediate actions for mitigating severe consequences are implemented without any evaluation of negative impacts due to implementation of suggested strategies. List of SCGs is in Table 3.

Table 3: List of Krško NPP TSC Severe Challenge Guidelines (SCGs)

TSC SCGs and associated SCST parameters		
Guideline	**Description**	**SCST parameter**
SCG-1	Mitigate fission product releases	Site releases and SFP level
SCG-2	Depressurize containment	Containment pressure
SCG-3	Control hydrogen flammability	Containment hydrogen below severe challenge
SCG-4	Control containment vacuum	Containment pressure

Mitigate Fission Product Releases (SCG-1)

The strategy for mitigating the fission product releases is called from the SCST. The purpose of reducing fission product releases is to protect the health and safety of the public. Actions in SCG-1 are identical to SAG-5, but they are implemented without any delay.

Concerning loss of SFP cooling and consequential decrease of SFP level, strategies are to refill SFP or spray water over the pool. Strategies for refilling the SFP and strategies for mitigating fission product releases from SFP are described in SAG-5 above.

Depressurize Containment (SCG-2)

The purpose of depressurizing the containment is to mitigate a severe challenge to the containment integrity due to high containment pressure. The consequence of not taking actions per this guideline will be containment failure leading to the uncontrolled release of high levels of fission products to the atmosphere. The only objective in SCG-2 is to reduce containment pressure far enough to mitigate a severe challenge. Guidance for reducing containment pressure to ambient conditions is then provided in SAG-6.

Containment over pressurization can be the result of a dynamic severe accident phenomenon (i.e., hydrogen burn) or a long-term pressure build-up due to steam or non-condensable gas build-up in the containment atmosphere. The dynamic severe accident phenomenon will cause pressure spikes that cannot be mitigated by the plant systems: Actions are considered in the DFC to prevent this dynamic severe accident phenomenon from occurring.

This guideline addresses actions to mitigate a severe challenge to the containment integrity due to high containment pressure. These actions are: depressurization of containment by RCFCs, injection of water by CI, and venting of containment by using various plant specific pathways.

Preferable way of depressurization is the use of sprays and fan coolers as described in SAG-6. However, the plant condition that caused the severe accident may also impact the availability of this equipment. Therefore, this guideline reviews the criteria necessary to operate the spray and fan coolers to determine if a containment heat sink can be established. Krško NPP has an alternative option if AC power is not available for CI, by the use of portable fire protection pumps or fire truck. There is a variety of water sources for alternative spray operation: fire truck tank, water treatment tanks, fire protection tank, condensate storage tanks, pretreated water tanks, condenser hotwell, city water, circulating water tunnel, and Sava river water, all equipped with fire protection connection points. Additional flow-paths are also provided: fire protections hoses connections at containment spray discharge lines and reactor coolant pumps fire protection spray nozzles.

Containment venting has been also identified as one of the potential recovery actions to mitigate a severe challenge due to the high containment pressure. Containment venting will reduce the pressure in the containment at the cost of releasing fission products from the containment atmosphere. Because of the fission product release, venting is considered the last option to prevent containment failure. During containment venting SCG-1 should not be implemented due to the higher priority to preserve containment as barrier for long-term mitigation of severe accident consequences. If a little water is available for spray strategy, the spray would be started for a few minutes before starting the vent strategy to scrub radioactive aerosols from containment atmosphere.

Applicable systems for containment venting identified in Krško NPP are: Reactor building hydrogen control system (Hydrogen purge) and Containment pressure relief system. If the control power for operation of venting valves is not available, there is a possibility for alternative power supply to specific bus with portable diesel generators. There is also a possibility to manually close ventilation dampers to minimize releases to auxiliary building if discharging fans are not available.

Control Hydrogen Flammability (SCG-3)

The purpose of controlling hydrogen flammability is to mitigate a severe challenge to the containment integrity due to a hydrogen burn. The consequence of not taking actions per this guideline will be containment failure leading to the uncontrolled release of high levels of fission products to the atmosphere.

Applicability of a control hydrogen flammability strategy is a function of the containment pressures, the containment temperature, hydrogen and steam concentration in containment.

If the plant is in the severe challenge for the containment integrity due hydrogen accumulation and potential hydrogen burn, then there are two major actions that can be taken to reduce the challenge: increase containment pressure or vent the containment. Increasing containment pressure can be done in two ways: stopping containment heat sinks and opening pressurizer PORVs. Stopping containment heat sinks would result in the build-up of steam in the containment atmosphere that would eventually inert the containment. Opening pressurizer PORVs would release steam to the containment atmosphere (assuming that the RCS has not completely depressurized), which would also inert the containment. Stopping heat sinks would result in containment pressurization as well.

Actions are performed in a way which minimizes the likelihood of creating an ignition source. For example, stopping the fan cooler motors may result in a spark. Instead, component cooling water should be isolated to the fan cooler unit to allow the containment to inert, and then the motor can be stopped. Also by isolating electrical power to containment charcoal cleanup fan motors, the potential for a hydrogen burn in containment will be reduced.

Venting the containment is another means of mitigating the containment challenge due to hydrogen combustion. Although venting the containment would not change the containment hydrogen concentration, it would reduce the mass of hydrogen in containment, which would in turn reduce the amount of energy released to the containment during a hydrogen burn. Limitations are offsite doses considerations, and consulting with dose assessment team is necessary. Venting is considered as the last option for mitigating containment hydrogen.

Control Containment Vacuum (SCG-4)

The purpose of controlling containment vacuum is to mitigate a severe challenge to the containment integrity due to the strong vacuum in containment. Containment pressure less than the lower containment pressure design basis can result in a severe challenge to the containment structure via buckling of the containment liner. A vacuum in containment may occur when non-condensable gases are released from the containment either via an unisolated leak that is subsequently isolated or from containment venting. After the leak is isolated or venting is stopped, containment pressure would remain above atmospheric due to steam in the containment atmosphere. However, if a containment heat sink is started, the steam would be condensed and the containment pressure can decrease below the containment pressure that existed prior to the accident. The minimum containment pressure would be a function of the amount of non-condensable gases released from the containment atmosphere. If the containment pressure has reached the point where containment failure is possible, this guideline would direct the TSC to pressurize the containment by stopping containment heat sinks or opening pressurizer PORVs. Other methods of adding non-condensable gases to the containment atmosphere have also been identified as potential recovery strategies in this guideline. Because of the consequences associated with containment failure, no short-term negative impacts have been identified for pressurizing the containment during a severe challenge. SCG-4 has an objective to increase containment pressure far enough to mitigate a severe challenge. Once the severe challenge is mitigated, a long-term concern will be identified to maintain containment pressure above the value for which a severe challenge may occur. Actions that can be taken to mitigate containment challenge due to vacuum are relatively simple and do not require any major decisions. Actions can be divided into two separate categories: short-term and long-term mitigation actions. Short-term mitigation actions are related to increasing the steam in containment, which can be done either by stopping containment heat sinks or opening pressurizer PORVs. These actions are considered short-term because they would mitigate the challenge rather quickly once they are performed, although they would not prevent the problem from occurring again later in the recovery. Long-term actions are related to increasing the non-condensable gases in containment, which can be done by establishing instrument air to containment, establishing nitrogen to the accumulators, etc. These actions are considered long-term because they would take some time to mitigate the challenge, although they would prevent the problem from occurring again later in the recovery.

TSC long-term monitoring (SAEG-1)

This guideline provides information to the TSC to monitor the long-term concerns associated with the implementation of strategies contained in the SAGs and the SCGs. Specifically, the information contained in this guideline relates to:

- Actions which must be taken after a strategy is implemented to ensure that the strategy can be continued in the long term,
- Actions which must be taken to ensure that a function can be continued in the long term, for systems functioning prior to entry into the SAMG,
- The potential for primary recovery methods to become available after an alternative recovery method has been implemented.

SAMG termination (SAEG-2)

This guideline is used after the plant is declared to be in a controlled and stable state using TSC diagnostic flowchart (i.e., core temperatures, reactor vessel level, site releases, containment pressure, containment hydrogen, SFP temperature are below limits and in acceptable state). It provides information for the TSC that is important to supplement recovery actions after the use of the SAMG is discontinued. Specifically, the information contained in this guideline relates to:

- Plant conditions that may prohibit recovery actions,
- Special conditions for long-term monitoring as a result of strategies that have been implemented and are continuing after the time the SAMG is terminated,
- High radiation concerns that should be taken into account in the recovery actions.

6.2.2 Accident management measures and installation design features for protecting containment integrity after occurrence of fuel damage

6.2.2.1 Management of hydrogen risks (inside and outside the containment)

6.2.2.1.1 Design, operation and organization provisions

Krško NPP containment design includes systems that are intended to mitigate and monitor potential hydrogen accumulation in containment following a design basis LOCA event. These are the Reactor building hydrogen control system and the Hydrogen monitoring system. The first system consists of two redundant electric hydrogen recombiners, located inside the containment and two redundant hydrogen purge ventilation systems. The hydrogen monitoring system uses an in-containment sensor arrangement and provides a means for measuring the containment hydrogen concentration. Systems comply with appropriate US codes and standards for design basis safety related equipment.

Hydrogen control during severe accidents requires a different approach from that for design basis events, primarily due to the more rapid generation and larger masses of hydrogen that can occur. Severe accident hydrogen control is defined in the SAMG, and uses existing equipment, including the hydrogen recombiners, hydrogen monitoring and purge system when appropriate, to perform strategies defined in the appropriate SAG/SCGs. This is described below.

In SAMGs, containment hydrogen concentration is recognized as a potential threat to the containment integrity and is therefore monitored and appropriate guidelines are entered if the concentration exceeds predetermined setpoint.

Immediate challenge to the containment fission product boundary is detected if containment hydrogen is above severe challenge, based on containment pressure and hydrogen concentration, also taking into account whether containment had previously been vented and whether core-concrete interaction had occurred. The SCG that controls containment flammability was developed to help mitigate this challenge. TSC is directed to isolate potential ignition sources (recombiners, ventilation motors) in the containment and to try to establish steam-inert atmosphere by shutting-off containment heat sinks and adding steam to containment atmosphere. If these actions were unsuccessful in reducing hydrogen severe challenge, TSC would consider venting the containment to the atmosphere using Hydrogen purge system (described above) or Containment pressure relief system, which is normally used for containment leak rate test. In order to assess whether the hydrogen in the containment atmosphere is flammable and/or there is a severe challenge to the containment, computational aid is included in SAMGs. This aid contains diagrams, which enable TSC to evaluate potential for hydrogen combustion based on the knowledge of the containment pressure and hydrogen concentration (additionally, based on previous recovery actions, hydrogen concentration can be assumed), also taking into account whether containment had previously been vented and whether core-concrete interaction had occurred.

Even if containment integrity is not jeopardized by the hydrogen burn, the fact that global hydrogen burn could occur (containment hydrogen concentration in dry air exceeds 4%) will direct TSC to enter the appropriate SAG, which provides guidelines for reducing hydrogen concentration. Using the computational aid, TSC estimates hydrogen impact to the containment, and is directed to initiate appropriate recovery actions.

Potential hydrogen escape route from the containment atmosphere and accumulation in other buildings than containment is through radiation monitoring penetration, as that line normally takes sample of the containment atmosphere and recirculates it back to the containment. The line contains motor-operated isolation valves both inside containment and in the auxiliary building, which normally close on containment ventilation isolation signal, but include handwheel and can thus be closed manually in case AC power is not available.

Hydrogen accumulation is possible in annulus (i.e. space between containment vessel and concrete shield building) in case containment vessel fails and annulus negative pressure ventilation is inoperable. There is no hydrogen monitoring system in annulus area.

Another potential source of hydrogen is located outside containment. If SFP cooling is lost, water in the pool would heat to boiling point and start to evaporate. With time, water level would drop below the top of the stored fuel elements and they would start to overheat. Eventually, cladding temperature would reach temperatures where zirc-water reaction is possible resulting in generation of more heat and producing hydrogen.

To prevent these challenges, temperature and level of the SFP are monitored in severe accident management guidelines. Both high temperature (close to boiling) and low level of SFP water would direct TSC to use the guidelines where mitigating actions are listed, consisting basically of different methods of adding water to SFP.

Water is normally added to the SFP from RWST or reactor makeup water storage tank. Provisions are also made on the SFP skimmer system with universal firefighting connections, through which gasoline-driven firefighting pumps can pump water from virtually any available tank on the plant, as well as from the Sava river. In addition to that, water can be sprayed over the pool, using the same pumps and firefighting nozzles, also located onsite.

6.2.2.1.2 Risks of cliff edge effects and deadlines

Cliff edge effect concerning hydrogen risk inside containment is recognized as potential accumulation of hydrogen gas in concentrations exceeding explosive mixture, where hydrogen detonation is almost certain to occur, plus elevated containment pressure, resulting in pressure peak that represents severe challenge to the containment (i.e. 5% possibility for containment failure).

In accordance with the Krško NPP plant-specific containment hydrogen distribution study, the results indicate that the duration of large hydrogen concentration peaks is short and it is coincident with high volume fractions of steam making them inert from combustion point of view. The containment dome is well mixed and there is no significant stratification. The assumptions used in the analysis were very conservative and no active strategies such as SAMG were modeled in order to assess limiting hydrogen concentrations. As it was concluded in the study, given large dilution volumes of the containment and its high ultimate load capability, hydrogen can be effectively handled using available means for hydrogen control and using existing SAMG.

Therefore no cliff edge effects or deadlines were recognized concerning hydrogen-induced threat to the containment fission product boundary.

In the SFP, cliff edge effect is recognized when water level in the SFP drops to the top of the fuel assemblies (uncovery of the top of the fuel assemblies is not expected to result in immediate overheating of the fuel elements as steam is expected to take away some heat; on the other hand, complete loss of shield will raise doses in the FHB dramatically, thus making personnel access very problematic). This margin in time depends on total heat power of the fuel elements currently stored in the pool. If bounding case is considered, where core has just been unloaded from the reactor after 18-months cycle, it would take approximately 3.2 days to uncover fuel elements.

In the SFP if sprays are ineffective in cooling the spent fuel, overheating of the fuel assemblies would be expected when water level drops to 20% of fuel assembly high within racks.

From then on fuel degradation is expected yielding exothermic zirconium oxidation and hydrogen generation. Time to reach the limit depends on total heat power of the fuel elements stored in the pool. If bounding case is considered, taking into account maximum decay heat load with core unloaded from the reactor immediately after 18-months fuel cycle, it would take approximately 6 days to that point.

6.2.2.1.3 Adequacy of the existing management measures and possible additional provisions

As it was concluded in the plant-specific study of the hydrogen distribution in the containment, given large dilution volumes of the containment and its high ultimate load capability, hydrogen can be effectively handled using available means for hydrogen control and using existing SAMG.

Concerning loss of SFP cooling, AOPs, EOPs and SAMGs direct TSC to address insufficient SFP cooling or decreasing SFP level by entering specific guidelines, where strategies to refill and spray over the pool are available. As an addition to the SFP filling methods, which are normally used by the operators, provisions are made with universal firefighting connections that allow pumping water into SFP with gasoline-driven firefighting pumps, stored onsite. These pumps can take suction from virtually any available tank in the plant, as well as from the Sava river. Additionally, water can be sprayed over the pool with these pumps.

6.2.2.2 Prevention of containment overpressure

6.2.2.2.1 Design, operation and organization provisions

There are three methods for decreasing the pressure in the containment: heat removal via component cooling water with RCFCs, containment spraying (or adding cold water to the containment) and containment venting.

Krško NPP has large, dry containment. During normal operation, pressure in the containment is kept slightly above the atmospheric pressure. In a severe accident, accumulation of steam and other gases in the containment, large enough to challenge the integrity of the containment due to the high pressure, creates an immediate challenge that is addressed in a specific SCG (specifically above 0.5 MPa). This guideline provides a systematic decision making path for reducing containment pressure to prevent containment failure. The objectives of the decision process incorporated into the guideline are to determine available means of depressurizing the containment, determine the preferred means of depressurizing the containment, determine if the challenge is being mitigated, and determine potential long-term concerns associated with depressurizing the containment.

In this guideline, preferred and most effective method of depressurizing the containment in terms of permanent heat removal from the containment is by establishing containment heat sink by operating RCFCs. Containment spraying is also very effective in reducing containment pressure by condensing steam and cooling hot gases in the containment atmosphere via heatup of the sprayed water droplets. Also, sprayed water tends to bind airborne fission products thus reducing radioactivity of the containment atmosphere.

If none of the above methods are available, provisions are made on containment spray piping for portable gasoline-powered firefighting pumps (two stored onsite, including fuel for a 3-day operation) that can pump virtually any available water on the plant (including Krško NPP's ultimate heat sink, the Sava river) into the containment via spray nozzles.

Least desired method of depressurizing the containment is by venting the containment, thus deliberately releasing fission products to the atmosphere. Although this method results in fission products release, controlled and limited release that can be terminated when containment pressure falls below dangerous levels is considered to be far better option than letting containment pressure rise until containment eventually fails, resulting in an uncontrolled and probably non-isolable fission products release path via containment break.

Containment venting is performed via Reactor building hydrogen control ventilation described in the previous chapter or Containment pressure relief system, which is normally used for containment leak rate test. Containment atmosphere is discharged through SFP exhaust ventilation, passing through HEPA-charcoal-HEPA filtering system where considerable amount of fission products would be deposited and thus prevented from escaping to atmosphere.

Given the pressure difference between the containment and the outside atmosphere is around 5 bars when executing this guidance, it will be more than sufficient to create a flow of air through ventilation ducts and filters even without ventilation fans operating. Valves in the lineup are fail-open, except for containment isolation valves which need power to open.

Since provisions were made on the plant instrument air system with quick-connectors for connecting mobile diesel-driven air compressor (two stored onsite with fuel for 3 days), air can be provided to the instrument air piping inside and outside containment, even in the case of station blackout and/or loss of ultimate heat sink. Also, additional mobile diesel generators are available on-site as explained in section 5.1.2.1, which can provide control power. Containment pressure relief system has containment isolation fail-closed valves that are air-operated and can, consequently, be opened under most beyond-design bases accidents.

Maintaining containment pressure as low as possible reduces the potential for containment failure as a result of pressure spikes that can occur due to certain severe accident phenomena (e.g. hydrogen deflagration). Containment pressure at near ambient conditions is a criterion for exiting SAMG. If containment pressure is above ambient, TSC is directed to a guideline that provides systematic decision making path for establishing a containment heat sink following core damage, in order to control the containment conditions, i.e. pressure and temperature. The equipment available is RCFCs or containment spray, as described in the previous paragraphs. Additionally, provisions were made on fire protection lines that are normally used to put out fire on RCPs, enabling mobile firefighting equipment to inject water into the containment through those lines. As a source of water, provisions exist on virtually all tanks containing relatively large amounts of water on the plant for portable firefighting pumps to take suction from (two gasoline-powered pumps stored on-site with enough fuel for a 3-day operation). Also, the Sava river water can be pumped with the diesel-powered submersible pump stored onsite. These pumps are completely self-sufficient, thus being able to operate under station blackout conditions.

6.2.2.2.2 Risks of cliff edge effects and deadlines

Cliff edge effect is defined as pressure at which containment fails. Generally, over-pressurization causes increased leakage rates or outright failure of the welds at containment penetrations or between containment liner plates. An overpressure challenge to containment is caused by the partial pressure of steam in the containment atmosphere, combined with any partial pressure contributed by non-condensable gases. In general, this type of internal overpressure is not expected to cause containment failure until containment pressure exceeds its design value by a factor of 2 or higher.

Potentially failure of containment (due to the over pressurization) can be significantly delayed by partial flooding of the containment by opening the gravity line from RWST. On the other hand the failure of containment could be prevented by spraying the containment with alternative portable fire pump (i.e. fire truck). By doing this containment pressure will remain below severe challenge for at least 7 days.

6.2.2.2.3 Adequacy of the existing management measures and possible additional provisions

Plant-specific analysis of various severe accidents that lead to core damage, made using accident analysis codes, showed that any of the methods available to mitigate containment high pressure severe challenge, i.e. containment recirculation fans, containment spray, flooding the containment or containment venting, would succeed in controlling the containment pressure below dangerous levels. Indeed, pumping water to containment at relatively small rates has proven to prevent or mitigate core-concrete interaction, thus minimizing production of non-condensable gases and steam that raises containment pressure. On the other hand, if the core becomes ex-vessel, adding cold water to the containment (together with heat dissipation on the containment wall) fairly compensates the generation of steam by the molten core-water reaction, keeping the containment pressure well below severe challenge for more than 7 days.

Furthermore, additional mobile equipment is stored onsite. This equipment consists of completely self-sustained diesel- or gasoline-driven pumps, compressors, generators, firewater and hydraulic piping, plus the fuel for 3 days of operation. Additionally, provisions with universal quick connectors exist on containment spray, reactor coolant pump fire protection, instrument air piping and virtually all plant’s tanks containing significant amounts of water, as well as ability to pump the Sava river water. With the personnel available non-stop on-site, namely shift crew and fire fighters, this equipment can be up and running and delivering water, air and electricity to the systems in the matter of few hours. With this equipment TSC can respond to the containment severe challenge even in the event of station blackout.

6.2.2.3 Prevention of recriticality

6.2.2.3.1 Design, operation and organization provisions

Borated water is stored in the RWST and two Boric acid tanks.

RWST, by design, contains enough water to ensure that sufficient water is available in containment to permit recirculation cooling flow to the core; boron concentration in the tank is kept within limits that ensure that the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control and shutdown rods out.

RWST serves as a suction source for ECCS pumps in injection mode. Provisions with universal firefighting connections were made for refilling the tank with portable firefighting equipment stored onsite.

Boric acid tanks contain enough borated water to ensure shutdown margin in all operating modes. Provisions were made on tanks with universal firefighting connections for the suction of mobile firefighting pumps stored onsite.

6.2.2.3.2 Risks of cliff edge effects and deadlines

Establishment of conditions in the reactor core that lead to recriticality and resulting in significant increase in heat generation is recognized as cliff edge effect.

No cliff edge effects related to core recriticality after core damage had occurred were recognized, as described in the following chapter.

6.2.2.3.3 Adequacy of the existing management measures and possible additional provisions

Although desirable, subcriticality is not a requisite condition for a controlled, stable in-vessel core. It is of concern only from the perspective of being able to remove all heat produced by the core.

The concern about potential recriticality of a damaged core comes from a conservatively postulated core condition. In a severely overheated core, control material could be the first material to relocate out of the core region. The situation postulated is that a large fraction of the control material is gone, but all or most of the fuel is remaining. The injection of unborated water into such a situation could improve neutron moderation to the point of achieving recriticality.

However, a damaged core has limited ability to return to a critical condition. The optimum reactivity configuration for the core is its normal geometrical array, which gives optimum moderator/fuel ratio. Melting or fragmentation of the fuel (compaction/consolidation) disturbs this optimum geometry. It reduces the moderator/fuel ratio, making the core less reactive. Thus, as core damage worsens, the core becomes less reactive because of geometry compaction, which reduces local moderator/fuel ratios, requiring less negative reactivity in the form of control rods or soluble boron in order to maintain subcriticality.

If a damaged core does return to a critical condition, power level would be limited. The void coefficient of reactivity is very effective at limiting power, regardless of any of the following conditions:

- Whether the injection water is borated or unborated,
- Whether the core is rodded or unrodded (control material is still present or has already relocated out of the core),
- Whether the accident occurs early or late in the fuel cycle (late in the fuel cycle, fission product poisons alone are sufficient to prevent return to criticality),
- Whether burnable poisons are present or absent.

The time response of the void coefficient depends upon the thermal response of the fuel, which is fast enough to compensate for positive reactivity addition from the reflood injection flow rates that can be achieved in a pressurized water reactor. The power level that completely vaporizes all of the injected flow bounds the power to which the core could return. The NSSS has sufficient relief capacity to accommodate the continuous steaming that would result from a stable power level.

Injection of unborated water can lead to a controlled and stable core state. The core might return to power, but only at a very low level, which is a function of the injection flow into the core. One of two outcomes would occur:

- The core would remain in a stable, but critical, state until borated water sources can be put in service; or
- The core would continue to degrade, resulting in a change in core geometry, which leads to a subcritical state.

Therefore, any available water should be injected into the core to achieve core cooling, eventually bringing the core to controlled and stable state.

Return to criticality is not a concern for ex-vessel core material. Compaction does not allow sufficient neutron moderation for criticality under any ex-vessel configuration. The portion of the core remaining in-vessel has greatly expanded geometry, precluding criticality for all possible configurations.

6.2.2.4 Prevention of basement melt-through: retention of the corium in the pressure vessel

6.2.2.4.1 Design, operation and organization provisions

See description in section 6.2.2.5.1.

6.2.2.4.2 Risks of cliff edge effects and deadlines

Reactor vessel failure is recognized as a cliff edge effect.

In the case that the vessel failure could not be prevented the best method to avoid MCCI is to flood the containment basement and rector cavity with water before reactor pressure vessel (RPV) fails.

This is assured by wet cavity design and accident management measures by procedure SACRG-1 which gives the instructions to the operators early before entering SAMG's DFC for flooding the containment by gravity line from RWST or by spraying the containment. Therefore no further cliff edge effects or deadlines were recognized concerning the basement melt-through due to the RPV failure.

6.2.2.4.3 Adequacy of the existing management measures and possible additional provisions

The best way for preventing reactor vessel failure is to quench and cool the core. When it comes to core injection, adding any type of water at any rate generally mitigates core damage progression. Severe accident guideline concerning core cooling is used by TSC in order to choose all available strategies for RCS injection, as well as to depressurize RCS to maximize RCS injection capability.

Limited plant design core injection capabilities are probably the reason for core damage at the first place. As an alternative, self-cooled positive displacement pump can be powered from auxiliary diesel generators, described in section 6.2.3.1. The pump can provide limited core injection capability from RWST or boric acid tanks.

Additional injection capabilities are possible with provision, made in the form of the spool piece, which can be mounted to the suction of the safety injection pump. Spool piece has universal firefighting connection, which allows personnel to inject water into RCS with mobile firefighting pumps, stored onsite.

If injection is not successful and relocation to the lower head occurs, RPV failure may result. Measures are taken before this would occur to inject water to the containment. One RWST tank volume would partially submerge the lower head. This measure may delay vessel failure but is unlikely to prevent it once relocation occurs. Since ECCS is constructed with recirculation capability, this water is in no way wasted for possible core cooling purposes if AC power is restored later in the accident.

The surest means of avoiding MCCI is to prevent failure of the RPV so that molten corium never has a chance to attack the containment concrete. However, it might not be possible to prevent vessel failure. The next-best means of avoiding MCCI is to flood the containment basement and the reactor cavity with water, if possible before the RPV fails.

6.2.2.5 Prevention of basement melt through: retention of the corium in the reactor pit

6.2.2.5.1 Design, operation and organization provisions

After vessel failure, basement melt-through can be prevented by ensuring that the core debris ex-vessel is water covered and cooled.

Krško NPP has wet reactor cavity, i.e. cavity is connected to containment sump with a 4-inch pipe. Consequently, flooding the containment will also flood reactor cavity. There are several methods available to inject water into containment:

- containment spray,
- RWST gravity drain,
- fire protection pipes for reactor coolant pumps,
- vacuum relief pipes.

Water can be injected to containment via containment spray from RWST with containment spray pumps. Provisions with universal firefighting connectors exist on spray lines, enabling portable firefighting pumps stored onsite to inject water into containment from virtually any water tank or the

Sava river. This method of injecting water is independent of the operability of the plant's system and can be performed even under station blackout.

RWST gravity drain lineup can be established by manipulating at least 2 motor-operated valves. Provisions were made on electrical feed for these motor-operated valves (in the form of standard 3-phase socket, mounted on the wall) so that they can be powered from mobile diesel generators, located onsite, enabling establishing RWST gravity drain to containment under station blackout.

Fire protection pipes for reactor coolant pumps also offer significant capability for injecting water into containment. For that purpose, three installed fire protection pumps can be used:

- flushing pump,
- high-capacity electric-driven pump,
- high-capacity diesel driven pump,

with first pump taking suction from fire protection tank, and the latter two taking suction from ESW intake structure, i.e. the Sava river. Provisions were made on the fire protection pipes for RCPs with universal firefighting connections, allowing pumping water to the containment with portable firefighting pumps stored onsite.

A spool piece with universal firefighting connection is stored in auxiliary building, which can be mounted on the containment vacuum relief pipes. Once installed, the connection is water-tight and can be used to pump water into containment with mobile firefighting pumps, stored onsite. To establish the lineup, two motor-operated valves have to be opened. Provisions were made on electrical feed for these valves (in the form of standard 3-phase socket, mounted on the wall) so that they can be powered from mobile diesel generators, located on-site, thus enabling manipulation of the valves under station blackout.

Severe accident guidelines with incorporated decision process enable TSC to determine available containment injection methods and deploy equipment and personnel to apply selected methods.

6.2.2.5.2 Risks of cliff edge effects and deadlines

Containment failure due to basement melt-through is recognized as a cliff edge effect.

Prevention of molten core-concrete interaction (MCCI) serves a double purpose: prevention of buildup of non-condensable gases in the containment atmosphere, and prevention of basement melt-through. In addition, prevention of MCCI avoids generation of potentially large quantities of hydrogen during the ex-vessel phase.

In the case that the vessel failure could not be prevented the best method to avoid MCCI is to flood the containment basement and rector cavity with water before RPV fails.

This is assured by wet cavity design and accident management measures by procedure SACRG-1 which is given the instructions to the operators early before entering SAMG's DFC for flooding the containment by gravity line from RWST or by spraying the containment. Therefore no cliff edge effects or deadlines were recognized concerning the basement melt-through due to the RPV failure.

Corium would also ablate the concrete, effectively attacking 4.5 m thick concrete slab below the reactor vessel. Since MCCI is end of thermal reaction, it consumes corium's decay heat, thus slowing down ablation process with time. Plant-specific analyses showed that corium would not reach the containment basement within 7 days.

6.2.2.5.3 Adequacy of the existing management measures and possible additional provisions

Core-concrete interactions occur when core material becomes ex-vessel and cannot be covered by an overlying water layer. An overlying water layer would cool the corium by transferring heat to the containment atmosphere. Krško NPP, originally designed with dry reactor cavity, made a modification to allow flooding the reactor cavity by connecting containment sump with the cavity. Basis for that modification were plant-specific analyses that showed advantages of protecting cavity floor with water against the corium, before reactor vessel fails. It also showed no adverse effect on reactor wall thermal shock.

Also, it is expected that adverse conditions, that would exist in the cavity once the corium is ex-vessel, would fail otherwise sealed door to the cavity, as well as reactor compartment ventilation duct, thus allowing more water to be admitted to the cavity floor.

Being recognized as principal cooling mechanism for cooling ex-vessel core debris, containment flooding strategies are available in the form of severe accident guidelines with incorporated decision process that enables TSC to determine available containment injection methods and deploy equipment and personnel to apply selected methods. Since it is so important to mitigate or prevent core-concrete interaction once corium is ex-vessel, containment flooding with RWST gravity drain is even included in plant's EOPs, as a response to the prolonged station blackout situation with significant damage to the onsite and offsite AC power sources.

In fact, it was shown by the plant-specific analyses that continually pumping relatively low amounts of water into the containment not only prevent MCCI, but also limit containment pressure rise, keeping it stable and well under containment severe challenge for more than 7 days.

Additionally to the ECCS pumps, provisions with universal firefighting connections exist on containment spray, fire protection piping for the reactor coolant pumps, and vacuum relief pipes, that enable injecting water into containment with gasoline- or diesel-powered firefighting pumps, stored on-site with fuel for a 3-day operation. Provisions exist, on virtually all plant tanks with substantial amount of water, that allow these pumps to take suction from, as well as pumping the Sava river water. Being completely self-sufficient, this method of injecting water into the containment can be set up with staff constantly available on the plant (shift crew and firefighters) within couple of hours, even in the case that all dedicated safety plant failed thus enabling containment flooding even under station blackout conditions.

6.2.3 Specific points

6.2.3.1 Need for and supply of electrical AC and DC power to equipment used for protecting containment integrity

By design, during normal operation, loads connected to the ESF buses and the power required for station auxiliaries are supplied from the generator through the unit transformers. Upon generator trip, power is supplied from 400 kV transmission grid system. Upon failure of the 400 kV source, power is transferred to the station auxiliary transformer, supplying power from 110 kV network. There is also a dedicated 110 kV power transmission line connecting Brestanica GPP and Krško, with the Brestanica's gas turbine »black start« ability (i.e. ability to start without external electric power). On complete loss of offsite power the ESF loads are automatically supplied by the safety-class standby diesel generators. If all AC power is lost, instrument and control power would be supplied to the control room by the two trains of safety-related batteries, designed with capacity of at least 4 hours.

In case all of the above plant design AC power supply fails, two auxiliary robust container diesel generators are stored onsite, with 600 kVA and 1000 kVA rated power each, and either being able to supply 400 kVA to the ESF 400 V bus (due to transmission cable limitations). Location of the generators is approximately 150 m away from and 2 m above the floor of the plant's safety-class diesel generators, thus making them unsusceptible to the common cause failure of the safety-class diesel generators. Both container diesel generators use jacket water heater and their batteries are constantly filled with charger. They are also capable of performing a cold-start from -20 °C. Enough fuel is stored onsite for diesel generators' 3-day operation.

It takes less than 1 hour for operating crew to connect the selected container diesel generator and start delivering electrical power to the 400 V safety bus.

The primary purpose of the container diesel generators is to power battery charger, which then provides power to one train of plant's AC and DC control power, thus enabling control room indications and controls, as well as control room lightning. Also, self-cooling positive displacement charging pump can be started, providing limited core injection capability. Depending on the momentary load of the transmission cable, there is some power left available to manipulate motor-operated valves as needed.

In addition to the two container diesel generators, three 150 kVA diesel generators are available on site. In case of unavailability of any other AC sources, these generators can be connected and power their respective motor control centers by simply plugging the ready cable into the socket, which is mounted on the wall near the motor control centers. The 150 kV generators are intended to provide quick alternative power supply to motor-operated valves, thus enabling plant's staff to manipulate them as required.

Water pumping capabilities on the plant are backed-up with two mobile gasoline-powered firefighting pumps, which are stored onsite. With provisions on the systems with universal firefighting connections installed, they can pump water from virtually any tank on the plant, and discharge water into the systems as directed by TSC in order to respond to containment challenges. Additionally, trailer-mounted diesel-powered submersible pump is stored onsite, which can be easily deployed to pump the Sava river water to the plant's systems. These pumps run free of any of the plant's systems, and can thus be successfully used even in the event of prolonged station blackout.

In case of the unavailability of the AC power and/or cooling water for the instrument air compressors, two diesel-powered high-capacity trailer-mounted compressors are stored onsite. Provisions on the instrument air system with quick-connectors allow them to supply compressed air to the IA system, both inside and outside of containment, even in case of prolonged station blackout and/or loss of ultimate heat sink. Compressed air can be used to manipulate air-operated valves in order to achieve important goals in protecting containment integrity, such as depressurize the RCS or open containment vent path.

Fuel, both diesel and gasoline, is stored onsite, in quantity that allows a 3-day operation on full load for all the above described equipment.

6.2.3.2 Adequacy and availability of the instrumentation

Instrumentation availability is a key to a successful implementation of severe accident guidelines. By design, safety-related batteries are intended to provide power to the instruments and indications in the control room for at least 4 hours following the complete loss of AC power. In operator's response to that situation, emergency procedures contain instructions to shed unnecessary loads in order to prolong the battery life, with the potential to double or even triple the design life. As described in the previous chapter, additional power-generating capabilities are installed onsite to provide instrumentation power for beyond design-bases prolonged loss of all AC power.

When addressing instrumentation adequacy during the severe accident, two problems arise: instrumentation survivability and adequate range.

MCR is equipped with safety related display instrumentation that enables operators to monitor the results of the ESF actions during design-bases accidents. In particular, a part of this instrumentation that is located inside the containment, i.e. transmitters, electrical cables etc., are designed to withstand prolonged influence to the environment of increased radiation, temperature, humidity, spray and pressure. However, during severe accident, conditions in the containment can deteriorate even beyond harsh conditions of the design basis accident (i.e. core at melting temperatures), rendering the safety-related instrumentation vastly inaccurate or even completely inoperable. This is also recognized in Krško NPP severe accident guidelines, and TSC is directed to observe all available indications related to the process, in order to decide whether to enter a specific guideline or confirm implementation of the chosen method. Alternative instrumentation list in the form of a table is available to the TSC to aid the personnel in identification of alternative instrumentation for the specific plant parameter.

During severe accident, many of the plant's parameters exceed by far their normal operating range, i.e. containment pressure, containment level, radiation, etc. Plant is equipped with wide-range instrumentation for all parameters that will be monitored during severe accident, mostly in the MCR and some locally, and severe accident management guidelines validation performed on the plant showed no deficiencies related to limited instrumentation range.

6.2.3.3 Availability and habitability of the control room

See chapter 6.1.2.2.

6.3 Spent fuel pool

Design provisions of spent fuel storage

Spent fuel pool design and configuration

Krško NPP spent fuel pool (SFP) and its auxiliary structures: Fuel transfer canal (FTC) and Cask loading area (CLA) for underwater nuclear fuel storage, handling and transport, form the Spent fuel pool cooling and cleanup system (SFPCCS). The walls of the SFPCCS are made of concrete, 1.83 m thick and are additionally fitted with 6 mm thick stainless steel liner plates to prevent leakage of water.

At Krško NPP site, SFP is located within Fuel handling building (FHB). FHB is an integral part of the auxiliary building and is a reinforced concrete structure that utilizes shear walls and beam and slab floor systems. It is designed in accordance with the seismic and other criteria for safety structures.

The FTC and CLA are connected to the SFP through passages. During normal operation the FTC and CLA are separated from SFP with stainless steel doors (gates) and could be empty.

Krško NPP SFP is designed to assure adequate safety under normal as well as under postulated accident conditions. SFP is designed to meet the requirements of 10CFR 20, in providing radiation shielding for operating personnel during fuel transfer and during storage of spent fuel.

The current Krško NPP SFP storage racks configuration consists of OLD stainless steel and NEW borated high density stainless steel storage racks. A total of 9 modules with 621 cells are provided in the old rack section that offers storage capacity for spent fuel plus one full core emergency unload. The new racks likewise comprise nine modules providing 1073 usable cells. This brings the total usable capacity of all racks installed to 1694 cells. OLD storage racks are administratively sectioned into two storage regions, i.e. Region I and Region II. They comprise of 621 low density cells, 21 of which are administratively prohibited.

All spent fuel racks are designed to withstand shipping, handling, normal operating loads (impact and dead loads of fuel assemblies) as well as SSE and OBE seismic loads meeting Seismic Category I requirements.

A SFP leak detection system is provided to monitor the integrity of the liners for the SFP, FTC and CLA.

Reactivity Control

SFP is divided into three distinctive and separate regions (Old Racks Region I, Old Racks Region II and New Racks Region II) that have different physics characteristics.

Criticality in the spent fuel storage racks is prevented by:

- physical separation of fuel assemblies (rack materials and dimensions),
- the presence of neutron absorbers (borated water in the pool and the use of borated stainless steel absorber sheets in the new spent fuel storage racks),
- administrative arrangement of spent fuel that specifies acceptable storage location based on fuel assembly reactivity (loading curve).

Note that reactivity equivalence principle takes into account decrease of reactivity due to fuel assembly operation history - burnup.

Under normal operation conditions as well as for postulated accident conditions, including pool boiling and optimum moderation, Old Racks Region I and Old Racks Region II calculated neutron multiplication factors do not exceed 0.95, including mechanical and calculations uncertainties. For Old Racks Region I as well as for Old Racks Region II usage the accident condition pool boiling at optimum moderation makes up the most reactive case. These configurations were analyzed with minimum required boron content of 2000 ppm to obtain operational loading curve (conservative approach).

Old Racks Region I area is strictly assigned for highly reactive fuel (including fresh fuel enriched to 5% U-235) in the 3 out of 4 storage configuration. In the Old Region II it is allowed to store the fuel assemblies that satisfy the respective loading curve.

During normal plant operation the Old Racks Region II are occupied typically with twice burned fuel with lower reactivity. The storage configuration in such a case is still subcritical in the case of hypothetical need to deliver non borated makeup water.

For the New Racks criticality safety is assured by geometrically safe configuration, the use of borated stainless steel absorber sheet and a procedure to verify that the reactivity equivalence curve provided in Plant's specific Technical Specifications is met. Criticality calculations conducted for the new racks demonstrate that there is a 95% probability that Keff would not exceed 0.95 at a 95% confidence level under normal and postulated accident conditions. In the New Racks Region II it is allowed to store the fuel assemblies that satisfy the respective loading curve. Pool boiling is not the issue due to decrease of reactivity with the waters density. This array is critically safe also for non borated (demineralized) water.

During the normal plant operation there is 2000 ppm boron in the pool water that ensures sufficient subcriticality in the case of multiple handling events (prevent postulated accidents).

Spent fuel pool cooling and cleanup system SFPCCS

The SFPCCS is designed to remove the decay heat generated by spent fuel assemblies stored in the SFP after it is removed from the reactor. A second function of the system is to maintain clarity and purity of the spent fuel cooling water and the refueling water.

To maintain spent fuel cooling water purification, a bypass circuit comprised of a mixed bed demineralizer and a filter is connected to the cooling loop. Water surface clarity is maintained by the function of the spent fuel skimmer system.

The SFPCCS has no emergency function during an accident. In the event of a failure of a SFP pump or heat exchanger, the second pump or other heat exchanger(s) will provide continued cooling of the stored spent fuel. This manually controlled system may be shutdown for limited periods of time for maintenance or replacement of malfunctioning components. The pool water volume is sufficiently large such that an extended period of time would be required for the pool water temperature to reach 100 °C if cooling were interrupted.

Protection against loss of shielding

The serious failure of SFPCC system would be complete loss of water in the SFP. To protect against the possibility, the SFP pump suction connections enter near the normal water level and the cooling water return line contains an anti-siphon hole. These design features assure that the SFP cannot be drained more than four feet below the normal water level (normal water level is approximately 7.8 m above the top of the stored spent fuel).

A makeup water system is provided to replace evaporative and leakage losses. It consists of supplying demineralized water from the demineralized water storage tank upon a low level signal from the SFP level instrumentation. Should the makeup water system not be operable, the secondary source of water would be from the reactor makeup water storage tank. The reactor makeup water storage tank is a Seismic Category I supply source.

As discussed above, system piping is arranged so that failure of any pipeline cannot drain the SFP below the water level required for radiation shielding. A depth of approximately 3.05 m of water over the top of the stored spent fuel assemblies is required to limit direct radiation to 2.5 mR/h (10CFR Part 20 limit for unrestricted access for plant personnel). It is estimated that in the case of at least 1 m water above stored spent fuel, the shielding for operators at SFP platform is still adequate.

Instrumentation and control

Temperature

Instrumentation is provided to measure the temperature of the water in the SFP, and to give local indication as well as annunciation at the main control board when normal temperatures are exceeded.

Instrumentation is also provided to give local indication of the temperature of the spent fuel cooling water as it leaves the heat exchanger.

There is also alternate equipment (different power supply from battery or generator, local indication, wide range) available onsite, which enables supervision of SFP temperature in the case of loss of AC power.

Level

Instrumentation is provided to give an alarm in the control room when the water level in the SFP reaches either the high or low level set points (15 cm above or 16 cm below the normal water level).

There is also alternate equipment (different power supply from battery or generator, local indication, wide range) available onsite, which enables supervision of SFP water level in the case of loss of SFP water without regard to its origin.

Walkdown

The operators perform walkdowns of the SFP area twice per shift to check the SFP system, temperature and level.

Fuel handling building SFP area charcoal cleaning exhaust system

During normal operation, refueling operation and under emergency conditions this system draws exhaust air continuously from the fuel handling building and a portion of the auxiliary building. Both filter plenums and any three of the four exhaust fans are required to operate.

6.3.1 Measures for managing the consequences of a loss of cooling function for the pool water

Guidance for action in case of losing cooling water pool is given in plant specific AOP and EOP. EOPs have higher priority than AOP, but AOP can be used for special guidance in parallel. In the case of decreasing level or increasing temperature of the SFP the operator will be warned by the activation of the MCR alarm: Initial corrective actions and guidelines will be given to the operators by the appropriate ARP which will further lead the operators to implement the recovery actions in accordance with the System operation procedure (SOP) section normal operation. However, if the deviations cannot be eliminated, then the operators take actions from the abnormal section. If the action prescribed in this section does not recover the deviation (in level or temperature) the procedure will give the instructions for transfer to the AOP Uncontrolled loss of reactor cavity or SFP.

The AOP recovery actions are structured in the following manner:

Heat removal from the SFP can be achieved basically in two ways: through SFP heat exchanger or through evaporation of water from SFP (or a combination of both). In the case of loss of cooling the only way is through evaporation of water with boiling. In this situation boron remains in the SFP and there is no concern about criticality.

Alternative means for establishing SFP water level and adequate cooling makeup water are provided to replace water lost through evaporation:

- Pumping water from water pretreatment tanks with portable fire pump to the system for purification of SFP water surface.
- Providing water from fire protection hydrant network to the system for purification of SFP water surface. This method requires pressurized fire protection hydrant network by installed diesel fire pump or by other portable diesel fire pump.
- Pumping water from pool near water pretreatment building with portable fire pump to the system for purification of SFP water surface.
- Pumping water from circulating water intake pool with submersible fire pump and fire track to the system for purification of SFP water surface.
- In the case of extensive SFP water leak to the AB sump provisions are made to establish the recirculation of this water back to the SFP by alternative mobile self powered pumps.

If water level in the SFP is decreasing even if makeup to the SFP is established, then operators are instructed to establish water spray over the spent fuel before the water drops below 3.05 m above the spent fuel elements. The priority of water sources is prescribed as follows: fire protection hydrant network, water pretreatment tanks, carbonate mud pool, circulating water intake and circulating water outlet pool.

Source of water can be provided from different tanks located at the plant, potable water, well water or water from the Sava river or any other water. Demineralized water or clean water without impurities would be preferred.

With the portable or mobile pumps with their own engines (independent from power source) water can be transported into the SFP. This could be done to the skimmer connection or directly with the use of fire protection hoses into the SFP. Provided there is balance of filling with water and evaporation there is no chance to lose the capability to cool the fuel and to lose the integrity of the spent fuel.

6.3.1.1 Before and after losing adequate shielding against radiation

Piping of the SFPCCS is arranged so that failure of any pipeline cannot drain the SFP below the water level required for radiation shielding. A depth of approximately 3.05 m of water over the top of the stored spent fuel assemblies is required to limit direct radiation to 25 μSv/h (2.5 mR/h) (10 CFR Part 20 limit for unrestricted access for plant personnel). It is estimated that in the case of at least 1 m water above stored spent fuel, the adequate shielding for operators at SFP platform is still adequate.

According to the plant specific study of the SFP water heatup and evaporation, the time to uncover fuel assemblies is 76 hours at maximum possible decay heat value 8.5 MW. It is expected that during this time the alternative strategies for SFP water inventory makeup, describing in EOP and in SAMGs would be implemented, which would enable long-term cooling of the SFP and provide adequate radiation shielding. The implementation of the corrective actions regarding the restoration of the SFP cooling capability are adequately described and addressed in the case that the plant is running on 100% power, in the case that reactor trip was initiated and in the case of the SBO. There is logical path of the procedural work flow during the restoration of the SFP cooling capability in all operation modes and plant condition on Figure 19.

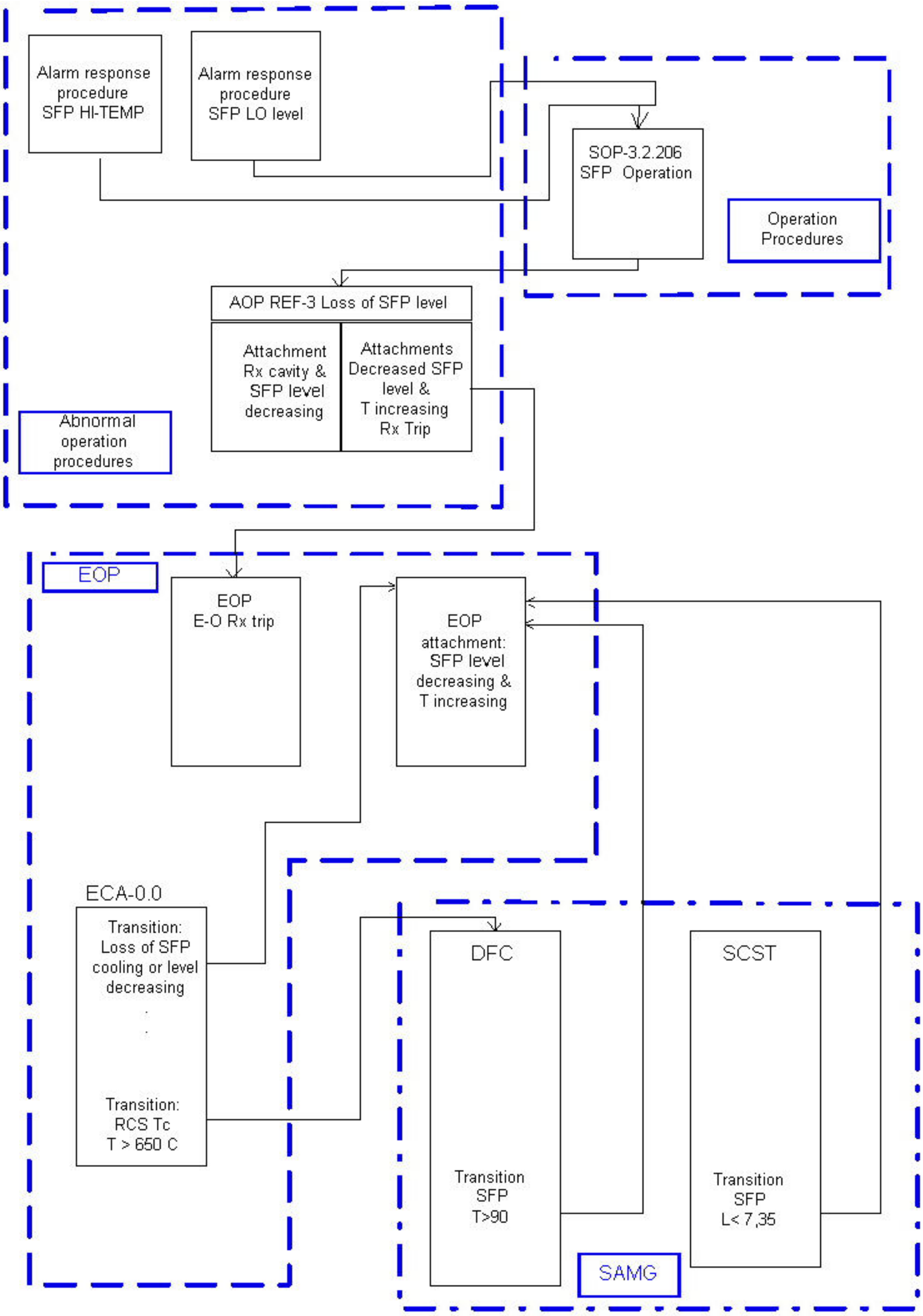


Figure 19: Procedural addressing of the SFP loss of cooling or level

The issue of the Loss of SFP cooling or level is also addressed regardless of the plant condition at the occurrence of this deviation. In the case that the plant is already faced with the SBO and the usage of the SAMG's procedures have been initiated the potential SFP issue is addressed by the existing diagnostic flow chart. Based on the plant radiation level or SFP level the operators (in fact the TSC is leading the usage of this sets of procedures) will be guided to make transition to the SAG-5 procedure, the purpose of which is reducing fission product release to protect the health and safety of the public and to establish the normal SFP cooling capability.

In the case of entry to the SAMG procedures to prevent challenges regarding SFP cooling, temperature of the SFP is monitored in this DFC and SCST check list guideline. High temperature (close to boiling) of SFP water will direct TSC to use mitigating actions, consisting basically of different methods of adding water to SFP. Water is normally added to the SFP from RWST or reactor makeup water storage tank. Provisions are also made on the SFP skimmer system with universal firefighting connections; through which portable fire protection pumps can pump water from virtually any available tank on the plant, as well as from the Sava river. In addition to that, water can be sprayed over the pool, using the same pumps and firefighting nozzles, also located onsite. Details are described in section 6.3.1. The list of methods for establishing the SFP normal conditions is similar to the instructions in AOPs, however the procedure also cover the potential SBO situation.

Diversity of these methods yields high level of confidence that level in the SFP would not drop below the top of the fuel elements and allow cladding temperatures to rise to levels where fuel cladding will degrade, fuel defragment causing severe radiological release.

SFP charcoal cleaning exhaust system function is to mitigate possible SFP radiological releases and removal of the released gases (also hydrogen).

6.3.1.2 Before and after uncovering of the top of fuel in the SFP

Mitigation and cooling restoration measures are the same as described in section 6.3.1.1.

As long as approximately 3.05 m above the top of stored spent fuel assemblies is maintained, radiological shielding is adequate and SFP is accessible for operator. It would take 47 hours to reach the 3.05 m water level above fuel elements under estimated maximum decay heat (8.5 MW).

It is estimated that in the case of at least 1 m water above stored spent fuel, the shielding for operators at SFP platform is still adequate.

6.3.1.3 Before and after fuel degradation in the fuel storage facility

If SFP cooling is lost, water in the pool would heat to boiling point and start to evaporate. With time and without any operator action, water level would drop below the top of the stored fuel elements and they would start to overheat.

Eventually, cladding temperature would reach temperatures above 650 °C where cladding will fail and gap release is expected causing the high radiological release of gases. When temperature increases Zircaloy cladding could reach temperatures at which the exothermic oxidation with oxygen in the atmosphere would become self-sustaining with resultant further damage of the cladding, fuel pellet relocation and pellet fission product release. Self sustained cladding oxidation (fire) of this type could occur at temperatures above 900 °C. The zirc water reaction is still possible but at even higher temperatures resulting in generation of more heat and producing some hydrogen.

To prevent these challenges, temperature and level of the SFP are monitored during the all different plant status and conditions regardless of the severness of the deviation (usage of abnormal set of procedures, with the reactor tripped, or in the SBO or SAMGs usage condition). Both high temperature (close to boiling) and low level of SFP water would direct operation to use the guidelines where mitigating actions are listed, consisting basically of different methods of adding water to SFP.

Water is normally added to the SFP from the RWST or reactor makeup water storage tank. Provisions are also made on the SFP skimmer system with universal firefighting connections, through which gasoline-driven firefighting pumps can pump water from virtually any available tank on the plant, as well as from the Sava river. In addition to that, water can be sprayed over the pool, using the same pumps and firefighting nozzles, also located onsite.

6.3.1.4 Risk of cliff edge effects and deadlines

The degradation of the fuel cladding and radiological releases starts with the uncovery of the fuel elements. The margin in time depends on total heat power of the fuel elements currently stored in the pool. If bounding case is considered, where core has just been unloaded from the reactor after an 18-month cycle, it would take more than 3.2 days to uncover fuel elements under estimated maximum decay heat (8.5 MW). Based on the more realistic (but still conservative) estimate that is valid for whole entire fuel cycle it would take about 11 days (2.5 MW) to uncover the fuel assemblies.

6.3.1.5 Adequacy of the existing management measures and possible additional provisions

As it was concluded in the plant-specific study of SFP water heat up and evaporation, loss of SFP cooling function could be handled using available means for supervision of SFP parameters (i.e. temperature and level) and using existing plant's specific abnormal or emergency operating procedures.

Concerning loss of SFP cooling, AOPs, EOPs and SAMGs direct operators/TSC to address insufficient SFP cooling or decreasing SFP level by entering specific guidelines, where strategies to refill and spray over the pool are available. As an addition to the methods for filling the SFP, which are normally used by the operators, provisions are made with universal firefighting connections that allow pumping water into SFP with gasoline-driven firefighting pumps, stored onsite. These pumps can take suction from virtually any available tank in the plant, as well as from the Sava river. Additionally, water can be sprayed over the pool with these pumps.

Diversity of these methods yields high level of confidence that level in the SFP would not drop below the top of the fuel elements and allow cladding temperatures to rise to levels where fuel cladding will degrade, fuel defragment causing severe radiological release.

6.3.2 Specific points

6.3.2.1 Adequacy and availability of the instrumentation

Instrumentation availability is a key to a successful implementation of severe accident guidelines. By design, safety-related batteries are intended to provide power to the instruments and indications in the control room for at least 4 hours following the complete loss of AC power.

EOPs instruct the operators to disconnect all non-essential DC loads. Based on plant specific best estimate DC study and with the actions of the operating crew to disconnect all non-essential DC loads the above mentioned 4 hours will prolong up and above 16 hours (train A) and 13.5 hours (train B).

However, with the multiplication of additional diesel generators (one fixed and five mobile), the instruction to strip all non-essential DC loads loses priority as the diesel generators ensure much longer availability of the batteries.

Establishing alternative power supply to the bus LD11 and to battery chargers from one of the two portable diesel generators will assure the long time availability of DC batteries and of 118 V AC instrumentation power supply (up to 72 hours since the fuel is stored at the plant for this time period, or even longer if the fuel would be supplied from outside of the plant).

Instrumentation is adequate during the severe accident, as long as water is present in the system.

This is also recognized in Krško NPP severe accident guidelines, and TSC is directed to observe all available indications related to the process, in order to decide whether to enter a specific guideline or confirm implementation of the chosen method. Alternative instrumentation list in the form of a table is available to the TSC to aid the personnel in identification of alternative instrumentation for the specific plant parameter. For the particular case (loss of SFP cooling function), this means existing alternate equipment for monitoring SFP parameters.

During severe accident, many of the plant's parameters exceed by far their normal operating range. The plant is equipped with wide-range instrumentation for all parameters that would be monitored during a severe accident, mostly in the MCR and some locally, and severe accident management guidelines validation performed on the plant showed no deficiencies related to limited instrumentation range.

In SFP, there are two type instruments related to supervision of the SFP parameters. The first one is for normal situation and care for adequate information about level and temperature, which are linked with process information system (PIS) and with main control board as an alarm when the water level in the SFP reaches either the high or low level set points (15 cm above or 16 cm below the normal water level) and high temperature in SFP (57 °C). These instruments have scale just for upper region of the SFP, which covers normal operation.

Early action on each alarm or malfunction in connection with SFP is guided by procedures for normal operation. Plant specific Technical Specifications declares requirements for level and temperature in SFP all the time when fuel assemblies are present in SFP. This represents enough information to determine status of SFP cooling and inventory during normal situation.

Alternate SFP level measurement covers all span (i.e. 12.12 m) and temperature at two different levels and indications at local panel. Signals are also connected to PIS system.

More challenging is to ensure indication during SBO and when level is dropping suddenly, constantly, due to evaporation or leakage. Indication panel is located in AB el. 115 and is accessible by stairs in AB building. Access is also possible from yard. There is a selector switch on the panel to transfer power supply to 24 V battery. With this power supply normal indication can be read, even if the power supply to PIS is lost. The purpose of this power supply is to ensure momentary indication (when you need the information) and after selecting this battery power supply. It is not intended for permanent use.

Thermo elements are RTD type, inserted into tube measurement for accurate temperature at the installed locations.

Level indication is ultrasonic type. This type of level indication can be inaccurate at high water temperature with steam with present at interface between water and air and can be oscillating which is an additional signal for SFP boiling. The indication would raise when boiling starts.

The internal video camera installed in the FHB connected to the plant computer network can be also useful for the supervising of the SFP area conditions.

Instrumentation that would not be useful until the SFP cooling start is established is as follows:

- TI1409 – Outlet manifold of the SFP heat exchangers;
- TE1415 – SFP heat exchanger SFAHSF01 outlet temperature;
- TE1416 – SFP heat exchanger SFAHSF02 outlet temperature;
- TE1417 – SFP heat exchanger SFAHSF03 outlet temperature.

Temperature transmitters and indicators TTI1415, TTI1416 & TTI1417 provide input signal for the PIS.

For online water temperature measurement, TE1421 is available.

TIS1400 measures temperature maintained in the SFP and gives a high alarm on the MCR board.

LS1401 measures water level in the SFP. It generates high and low alarms on the MCR board. For online water level measurement, LE1420 is available.

LI1401 is a measuring ruler, which measures water level in the SFP.

6.3.2.2 Availability an habitability of the control room

Habitability systems for the control room are designed so that habitability can be maintained under normal and accident conditions.

6.3.2.3 Potential for hydrogen accumulation

In the event of a rapid loss of cooling water the zirconium cladding of such overheated spent fuel can reach temperatures where it would burn, causing a spent fuel fire that could lead to fuel melting and a large release of radioactivity.

When fuel overheats in an accident, explosive hydrogen gas could be generated by the interaction of steam with metallic fuel cladding.

There is no on-line monitoring of hydrogen concentration in the area of the SFP.

7 General conclusion

7.1 Summary with potential areas for improvements and action plan

It can be concluded that:

1. Krško NPP is well designed and very robust regarding its capability to withstand well above design bases accelerations due to an earthquake. The following are main conclusions regarding seismic margin and cliff edge effects:
   - Conclusion regarding the seismic core damage margin

     Based on the seismic margin evaluation, taking into account the alternative means described in the EOPs and SAMGs, it is considered that seismic levels at which core damage would be likely are at PGA range of 0.8 g or higher. At these seismic levels, the critical induced sequence is seismic ATWS with SBO conditions. Seismic ATWS could, at seismic events of such a severity, be caused by a failure of control rods insertion due to degradation of fuel assemblies' geometry. Although the long term shutdown (sub-criticality) can still be achieved (boric acid transfer system), the critical function is ensuring the secondary heat sink in time. Following the seismic failure of CST, together with conditions of induced SBO and / or loss of ESW, the secondary heat sink would have to be provided by alternative means specified in the EOP ECA-0.0 Appendices. It is questionable, however, whether this can be implemented in time, considering the ATWS condition.

     Also, the liquefaction cannot be excluded which would potentially fail buried structures and / or equipment.

     Seismic capacities of structures related to primary or secondary pipe breaks (i.e. LOCAs or SLBs) are fairly above these levels.

   - Conclusion regarding the seismic margin for containment integrity

     Early releases

     Seismic events at which early radioactivity releases into the environment would be likely to occur are considered to be of PGA significantly exceeding 1 g.

     Late releases

     Seismic events at which late radioactivity releases into the environment would be likely to occur are considered to be of PGA in the range of 0.8 to 0.9 g or higher.

   - Conclusion regarding the SFP integrity – cliff edge effect

     For earthquake levels up to, approximately, 0.9 g, it is considered that the SFP integrity would not be challenged. Alternative strategies from EOP procedures and SAMGs are credited to provide the makeup water for the SFP inventory and, thus, prevent the FAs from overheating in the case of small leakages or loss of inventory during evaporation.

     Accordingly, for earthquakes in the range of PGA exceeding 0.9 g, gross structural failures of SFP cannot be excluded. For earthquakes of such intensity, fuel uncovers in the SFP are considered likely to occur.

   At the end, it needs to be pointed out that seismic events with PGA higher than 0.8 g were estimated to be very rare events at NEK site. Based on the revised PSHA and Seismic PRA, the return period for such an event is considered to be larger than 50000 years.

2. Reactor core damage, and hence, a challenge to containment, can be avoided at water flows of the Sava river significantly higher than 7100 m^3/s. Since the plant is shut down, the sequence development (to the point of core uncovery and fuel damage) would be slow and would enable the implementation of alternative methods described in the EOPs and SAMGs even with NPP plain flooded to a certain level. If the core is preserved, there would be no challenge to the containment. The most recent studies indicate that actual flooding would not start below 11000 m^3/s flow, which is much higher than a 10000-year return period flow (4790 m^3/s) or new revised PMF of 7081 m^3/s. Having in mind that 1000-yr and 10000-yr floods are estimated at 4040 m^3/s and 4790 m^3/s, respectively, it can be expected that the return period for the flood as large as 11000 m^3/s would lie in the range of 1E+06 yr or more.

3. In the case of Loss of the primary heat sink and/or loss of all AC, Krško NPP can:
   - Assure safe condition of the reactor for at least 7 days providing the water source for decay heat removal for turbine driven auxiliary feedwater pump. In case turbine driven auxiliary feedwater pump is not available, portable fire protection pumps can be used to supply water into the steam generator. These pumps have enough capacity to remove the decay heat from the core and to maintain the level in steam generator to provide natural circulation on the primary side. To assure AC power supply for instrumentation and some key AC powered equipment, Krško NPP has two portable DGs (1 MVA and 0.6 MVA) which can be connected to appropriate AC distribution network within 1 hour by operation crew onsite. In addition Krško NPP has also 5 portable engine driven generators on site to provide power to essential instrumentation.
   - Assure spent fuel pool cooling by adding cooling water with portable pumps. For the worse case (the entire core is unloaded to the spent fuel pool), the time available to establish water injection into the spent fuel pool is 47 hours (at that time water would drop to USAR limit (shielding) value of 3.05 m). It was calculated that the fuel would remain covered for 76 hours after event initiation. As indicated in respective chapters of this report, Krško NPP already implemented the majority of USNRC requirements related to B.5.b and portable equipment is onsite, already available for such purposes.

4. Krško NPP has in place upgraded EOP and SAMG procedures which also provide adequate instructions for mobile equipment available onsite.

5. Krško NPP has in place Radiological Emergency Response Plan (RERP) which is coordinated with the RERPs of Krško and Brežice municipalities (local RERPs), with the RERP of Posavje region and with the national RERP. The emergency preparedness planners on all levels coordinate their effort and activities regularly.
   - Onsite Emergency Response Organization (ERO) is established. The ERO intervention teams (including operators and security guards) can be exchanged during interventions and long-term emergency response. During the accident, the intervention staff is located in emergency response facilities (MCR, TSC and OSC), which are structured, equipped and organized to enable long-term habitability.
   - Offsite support and assistance to Krško NPP is provided by the local and other offsite support organizations. Contracts and letters of agreements have been developed and signed to delineate outside company/agency assistance and services. The contracts and letters of agreement are reviewed annually to reaffirm assistance and to verify communication channels.

6. Krško NPP has man forces, mobile equipment and resources to manage initial emergency response in case of a severe accident for an extended time - up to 24 hours without any offsite support and up to one week with no needs for additional heavy mobile equipment from offsite. The mobile equipment essential for managing a severe accident (SAME) according to EOPs and SAMGs strategies is stored onsite. The SAME is placed on safe locations to avoid impairments due to accidental conditions (earthquake, floods, fire etc.). Fuel is stored on-site for mobile equipment in the quantities for at least first 72 hours. The mechanical connections, power supplies, connection tools and other arrangements are prepared in advance on locations and on components of systems where SAME should be connected or applied to implement required severe accident management strategies. This enables preparation and implementation of severe accident management strategies only with shift crews under accident conditions in an effective manner after making a decision to implement the strategy.

   The SAME is included in Krško NPP equipment data base as an AE system and is regularly tested and maintained in accordance to plant maintenance procedures. Regular training and drills for shift personnel and other personnel in ERO responsible for implementing the severe accident strategies and handling with the SAME are conducted on an annual basis.

7. Krško NPP is implementing (projects under execution – completion in 2012) additional safety upgrades, in particular:
   - Third independent diesel generator with a safety bus, which can be connected to both existing safety buses.
   - Provision to connect mobile diesel generator of capacity 2000 kVA to switch gear of the third diesel generator.
   - Flood protection upgrade (increasing the level of the dikes upstream of the site on the left bank of the Sava river and of the tributary Potočnica), to keep the left Sava river bank dry even for flows beyond the PMF flood flow.

8. In addition to the above listed safety upgrades already being implemented, Krško NPP is in process of implementation of the following additional improvements to increase plant capability to withstand severe natural phenomena:
   - Acquiring (purchasing) onsite additional mobile diesel generator of capacity 2000 kVA.
   - Acquiring (purchasing) onsite additional pumping station to assure additional high capacity »portable water ring« around the plant - »HFS HydroSub 450 floating unit«.
   - Acquiring (purchasing) two additional high pressure mobile fire protection pumps which would be available for possibility to remove decay heat in early stage after reactor shutdown and depressurizing steam generators.
   - Installation of some additional quick connection points for mobile equipment.

7.2 Potential safety improvements and further work forecasted

As already mentioned in the introduction of the report, besides obligating the plant to perform the stress tests, the SNSA also issued a decision that requires from the plant to reassess the Severe Accident Management strategy, existing design measures and procedures and implement necessary long term safety improvements for prevention of severe accidents and mitigation of its consequences. This evaluation shall be finished by January 2012, while the action plan shall be completely implemented by the end of the year 2016.

In the scope of this reassessment it is expected that the plant will come up with several modifications that would further enhance the robustness of the power plant (with stress on the power sources, additional means of cooling and containment integrity), among others also the construction of a new backup control room.

Additional modification that would increase the reactor safety is the construction of the alternate ultimate heat sink. This modification is being considered in the siting process of the downstream HPP Brežice as a countermeasure to impacts that the construction and operation of the new hydro plant would have on the Krško NPP. The alternate UHS is in the design phase. It is supposed to be a safety class and seismically qualified structure and would be mostly independent of the Sava river (as oppose to the primary UHS).

In the meantime the 2nd periodic safety review is underway in the framework of which the plant's design and operation will be compared against the newest standards and best industry practice. Again, the site characteristics will be checked to see whether there have been changes which must be considered. Like the 1st PSR, which gave good results and helped with its recommendations and suggestions to increase plant's robustness and thus safety of the facility, it is also expected that the 2nd PSR will do the same.

Besides that the SNSA will take into consideration all suggestions or recommendations that may arise from the peer review of the stress tests, and incorporate them in the future actions to further enhance the nuclear and radiation safety of the Slovenian nuclear industry.

8 Appendix I: Technical Data

1 NUCLEAR POWER PLANT

Reactor thermal power	MW	1994
Gross electrical output	MW	727
Net electrical output	MW	696
Engineering minimum	MW	32
Heat consumption	kcal/kWh	2560
Thermal efficiency factor	%	36

2 REACTOR VESSEL

Chemical composition		H_2O
Additives		H_3BO_3
Number of cooling loops		2
Total mass flow	kg/s	9220
Pressure	MPa	15.41
Total volume	m^3	197
Temperature at reactor inlet	°C	287
Temperature at reactor outlet	°C	324

3 REACTOR COOLANT PUMPS

Number of pumps		2
Pump capacity	m^3/s	6.3
Design pressure	MPa	17.13
Design temperature	°C	343.3
Seal water injection	m^3/s	1.82
Seal water return	m^3/s	0.68
Cooling water flow	m^3/s	49.97
Motor power	kW	5222

5 PRESSURIZER

Design pressure	MPa	17.13
Design temperature	°C	360.0
Surge line nozzle diameter	m	0.35
Heatup rate of pressurizer using heaters only	°C/h	30.6
Internal volume	m^3	28.3

4 STEAM GENERATORS

Number of steam generators		2
Steam pressure at steam generator outlet	MPa	6.5
Design pressure, reactor coolant side	MPa	17.2
Design pressure, steam side	MPa	8.3

Feedwater temperature at steam generator inlet	°C	219.4
Total steam mass flow	kg/s	1088
Steam generator height	m	20.7
Steam generator weight	t	345
Number of U-tubes in a steam generator		5428
Total heat surface	m^2	7177
U-tube outside diameter	mm	19.05
U-tube Thickness	mm	1.09

6 NUCLEAR FUEL

Number of fuel assemblies		121
Number of fuel Rods per assembly		235
Fuel rod array in fuel assembly		16×16
Fuel rod length	m	3.658
Clad thickness	mm	0.572
Clad material		Zirlo
Fuel chemical composition		UO2
Pellet diameter	mm	8.192
Pellet height	mm	13.460
Total weight of nuclear fuel	t	48.7

7 CONTROL RODS

Number of control rod assemblies		33
Number of absorber rods per assembly		20
Total weight of control rod assembly	kg	53.07
Neutron absorber		Ag-In-Cd
Percentage composition	%	80-15-5
Diameter	mm	8.36
Density	g/cm^3	10.16
Clad thickness	mm	0.447
Clad material		SS 304

8 EMERGENCY CORE COOLING SYSTEM (ECCS)

SAFETY INJECTION SYSTEM (SI)		
Safety class		2
Seismic category		I
Number of pumps		2
Design pressure (gauge)	MPa	17.06
Design temperature	°C	148
Design flow rate	m^3/h	159
RESIDUAL HEAT REMOVAL SYSTEM (RHR)		
Safety class		2
Seismic category		I

Number of pumps		2
Design pressure (gauge)	MPa	4.12
Design temperature	°C	204
Design flow rate	m^3/h	511
ACCUMULATORS		
Safety class		2
Seismic category		I
Number of accumulators		2
Relief valve setpoint	MPa	5.49
Design temperature	°C	148
Normal pressure	MPa	5.17
Total volume	m^3	56.6
BORON INJECTION TANK		
Safety class		3
Seismic category		I
Number		1
Design pressure (gauge)	MPa	18.83
Normal pressure (gauge)	MPa	0.69
Total volume	m^3	3.4
Range of boric acid concentration	ppm	2550 – 3330
NORMAL OPERATING STATUS OF ECCS COMPONENTS FOR CORE COOLING		
Number of SI pumps operable		2
Number of RH pumps operable		2
Number of heat exchangers operable		2
Minimum refueling water storage tank volume	m^3	1250
Boron concentration in RWST	ppm	2550 – 3130
Boron concentration in accumulators	ppm	2550 – 3130
Number of accumulators		2
Minimum accumulator pressure	MPa	4.8

9 CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

Safety class		2
Seismic category		I
Seal water supply flow rate for each RCP	m^3/h	1.8
Seal water return flow rate for each RCP	m^3/h	0.68
Letdown flow, normal / maximum purification	m^3/h	17 / 27
Charging flow, normal / maximum purification	m^3/h	14.7 / 25
Temperature of letdown reactor coolant	°C	281.9 – 288.1
Temperature of charging flow to RCS	°C	270
Temperature of effluent directed to boron recycle system	°C	46

POSITIVE DISPLACEMENT PUMP		
Number		1
Design pressure	MPa	21.38
Design temperature	°C	121
Design flow	m^3/h	7.9
CENTRIFUGAL CHARGING PUMPS		
Number		2
Design pressure	MPa	21.38
Design temperature	°C	121
Design flow	m^3/h	36.3
BORIC ACID TRANSFER PUMPS		
Number		2
Design pressure	MPa	1.03
Design temperature	°C	121
Design flow	m^3/h	7.9 / 22.7

10 COMPONENT COOLING SYSTEM (CC)

Safety class		3
Seismic category		I
Number of loops		2
Number of pumps		3
Pump rated capacity	m^3/h	1828
Pump design pressure	MPa	1.38
Number of heat exchangers		2

11 ESSENTIAL SERVICE WATER SYSTEM (SW)

Safety class		3
Seismic category		I
Number of loops		2
Number of pumps		3
Pump rated capacity	m^3/h	2880
Pump design pressure	MPa	0.59
Design river water temperature	°C	0.6 – 26.7
Designed for operation with water level between	m.a.A.s.l.	147.85 – 156.60

12 AUXILIARY FEEDWATER SYSTEM (AF)

Safety class		3
Seismic category		I
MOTOR DRIVEN PUMPS		
Number of pumps		2
Rated capacity	m^3/h	84.1
Design pressure	MPa	12.41

TURBINE DRIVEN PUMP		
Number of pumps		1
Rated capacity	m^3/h	191
Design pressure	MPa	12.41
Steam inlet pressure, minimum / maximum	MPa	0.57 / 6.55

13 CONTAINMENT

Height	m	71
Inside diameter	m	32
Outside diameter	m	38
Steel shell test pressure	MPa	0.357

14 CONTAINMENT AIR RECIRCULATION AND COOLING SYSTEM

Safety class		2
Seismic category		I
CONTAINMENT SPRAY SYSTEM (CI)		
Number of loops		2
Design pressure (gauge)	MPa	2.41
Design temperature	°C	148
Design flow rate	m^3/h	270
Auto actuation on cont. HI-3 pressure signal	MPa	1.59
CONTAINMENT RECIRCULATION FANS (RCFC)		
Number of units		4
Capacity to remove heat of each unit during normal operation	MW	0.73
Capacity to remove heat of each unit during post-LOCA conditions	MW	17.0

15 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

Safety class		2
Seismic category		I
Number of electric hydrogen recombiners		2
Processing capacity (per unit)	m^3/min	2.83
Designed to sustain all normal loads as well as accident loads including SSE and pressure transients from a design basis LOCA		
Hydrogen control system (as a backup to recombiners)	Number of trains (2×100%)	2

16 FIRE PROTECTION SYSTEM (FP)

Safety class		NNS
Seismic category		-
Dedicated FP water tank volume	m^3	379
Jockey pump for pressure maintaining	m^3/h	4.54
Jockey backup pump	m^3/h	432

Primary FP water supply pump (electric) flow rate	m^3/h	568
Primary FP water supply pump (electric) discharge pressure	MPa	0.86
Primary FP water supply pump (diesel) flow rate	m^3/h	568
Primary FP water supply pump (diesel) discharge pressure	MPa	0.86

Rooms or areas with smoke detectors:

- the cable area above the ceiling of the main control room,
- the sub floor and above floor areas of the Computer Room,
- the CRDM Control Area on El. 115.55 m,
- Battery Rooms A and B on El. 100.3 m,
- Switchgear Rooms A and B on El. 107.62 m,
- cable spreading rooms on El. 104.87 m and 111.89 m of the Control Building,
- vertical cable chases near the Control Building,
- horizontal cable chases A and B on El. 97.62 m of the Auxiliary Building,
- electrical equipment rooms and Work Control Center on El. 115.55 m of the Turbine Building,
- switchgear area on El. 107.62 m of the Turbine Building,
- safety related pump rooms,
- safe shutdown panel locations,
- inside all main control room panels,
- fuel handling and storage areas,
- main control room ceiling,
- radwaste Building,
- drumming station room,
- Switchyard Auxiliary House,
- TP1, TP2 and TP3 Buildings,
- Pretreatment Building,
- Switchgear House,
- Secondary Chemical Laboratory Room on El. 100.3 m of the Turbine Building;

Each control panel for smoke detectors also has a manual fire alarm station associated with it so personnel can initiate a fire alarm signal to the Main Control Room (Ventilation Board). Manual fire alarm stations are also installed:

- at the entrance of IDDS Drumming Station Room,
- at the entrance of Switchyard Auxiliary House,
- in the Demineralised Water Tank area,
- in the west of Condensate Tank area,
- in the southeast of the Decontamination Building area,
- at the personnel entrance of the Pretreatment Building,
- at the entrance of TP3 Building,
- in the Cooling Tower 2 area,
- on the River Dam Left Bank Pier.

Pump running alarms and pump trouble alarms for the two fire pumps are also sent to the Main Control Room (Ventilation Board) for annunciation.

17 INSTRUMENT AND COMPRESSED AIR SYSTEMS (IA & CA)

Safety class		NNS
Seismic category		Non
INSTRUMENT AIR SYSTEM		
Number of loops		2
Number of compressors		3 × 100%
COMPRESSED AIR SYSTEM		
Number of loops		1
Number of compressors		1

Safety related systems, which require compressed air for their operation, are equipped with local safety class 3 control air subsystems, composed of accumulator tanks or bottles with necessary hardware.

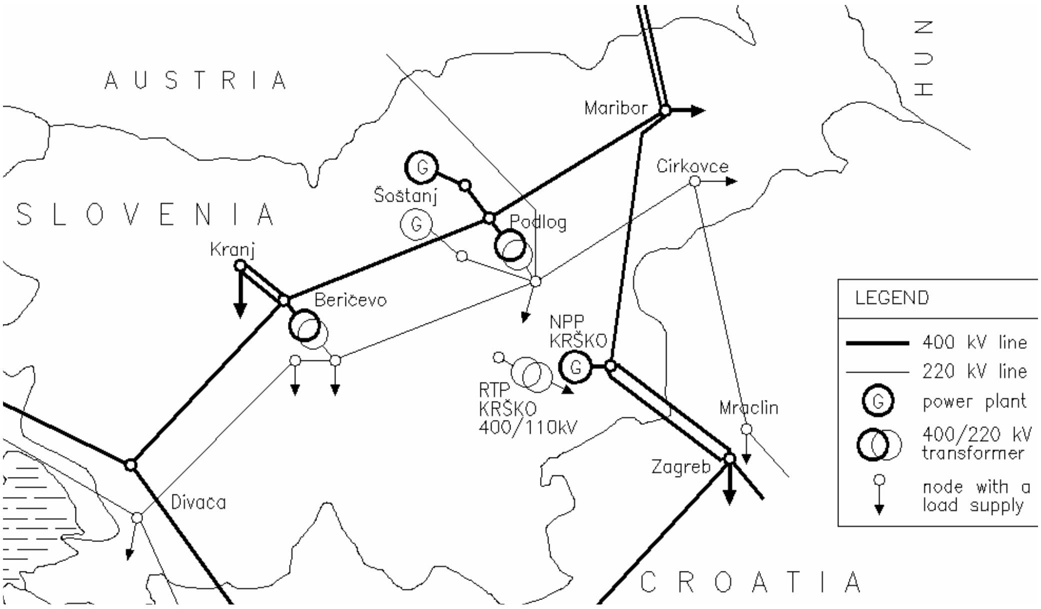
18 SPENT FUEL POOL (SFP)

Safety class		3
Seismic category		I
Spent fuel pool storage capacity (all)		1694
Spent fuel pool water volume (free volume)	m^3	1629
Nominal boron concentration	ppm	2000
STORAGE		
a. Design case 40% of a core load with 40 yrs. storage from previous refuelings		
Decay heat production	kW	3590
SFP temperature	°C	<51.8
SFP heatup rate assuming loss of cooling	°C/h	1.89
b. Maximum heat load case – 1694 elements stored (full rack plus complete unloading of the core)		
Decay heat production	kW	8360
SFP temperature	°C	73.5
SFP heatup rate assuming loss of cooling	°C/h	4.41
OLD SPENT FUEL STORAGE RACKS		
Number of cells		621
Rack material		Austenitic SS
NEW SPENT FUEL STORAGE RACKS		
Number of cells		1073
Rack material		Borated SS
SPENT FUEL POOL COOLING AND CLEANUP SYSTEM (SFPCCS)		
Number of SFP pumps		2
Design pressure	MPa	1.03
Design temperature	°C	93
Design flow	m^3/h	318
Number of refueling water purification pump		1

NEW FUEL STORAGE POOL		
Number of dry storage for new fuel elements		42
Rack material		Austenitic SS

19 ELECTRICAL POWER

OFFSITE POWER		
Number of 400 kV transmission circuit terminals		3
Number of 110 kV transmission circuit terminals (connected with gas-steam power plant Brestanica)		1
ONSITE POWER - AC		
Number of independent 1E 6.3 kV buses (powered also by emergency diesel generators)		2
Number of independent non-1E 6.3 kV buses		2
ONSITE POWER - DC		
Number of 1E 125 V DC system		2
Number of non-1E 220 V DC system		1



Connections to neighboring countries

20 DIESEL GENERATOR SYSTEM (DG)

Safety class		1E
Seismic category		I
Number of diesels		2
Rated voltage	V	6300
Rated power	kW	3500
30 min. short time rating	kW	4178
2000 hour rating	kW	3893

Day tank capacity (for each DG)	hours of operation	4
Oil storage capacity (for each DG)	days of operation	7
Time from SI start to rated speed	s	10

21 TURBINE GENERATOR

Maximum power	MW	727
Steam flow rate	kg/s	1090
Fresh steam inlet pressure	MPa	6.2
Fresh steam temperature	°C	275.5
Turbine speed	rad/s rotation/min	157 1500
Steam moisture at high-pressure turbine inlet	%	0.46
Condenser pressure	kPa	5.1
Average condensate temperature	°C	33.0
Number of feedwater pumps		3 × 50%
Generator rated power	MVA	813
Rated voltage	kV	21
Generator rated frequency	Hz	50
cos Ø	Ø	0.85
Regulated range	%	+10-7

22 SEVERE ACCIDENT MANAGEMENT EQUIPMENT (SAME)

Portable generator AE900AGR-001	0.4 kV / 5 kW
Portable generator AE900AGR-002	0.4 kV / 5 kW
Portable generator AE900AGR-003	0.23 kV / 2.6 kW
Portable generator AE900AGR-004	0.23 kV / 2.6 kW
Portable oil free compressor AE900CPR-001	1620 m^3/h / 1.03 MPa
Portable oil free compressor AE900CPR-002	1620 m^3/h / 1.03 MPa
Mobile diesel generator AE900DSL-001	0.4 kV / 600 kVA
Mobile diesel generator AE900DSL-002	0.4 kV / 1000 kVA
Mobile diesel generator AE900DSL-004	0.4 kV / 150 kVA
Mobile diesel generator AE900DSL-005	0.4 kV / 150 kVA
Mobile diesel generator AE900DSL-006	0.4 kV / 150 kVA
Portable fire protection pump AE900PMP-001	50 kW / 60 m^3/h / 1.5 MPa
Portable fire protection pump AE900PMP-002	50 kW / 60 m^3/h / 1.5 MPa
Submersible pump AE900PMP-003	2.8 kW / 7 m^3/h / 0.2 MPa
Submersible pump AE900PMP-004	2.8 kW / 7 m^3/h / 0.2 MPa
Submersible pump AE900PMP-005	2.8 kW / 7 m^3/h / 0.2 MPa
Submersible pump AE900PMP-006	2.8 kW / 7 m^3/h / 0.2 MPa

Trailer with HS60* HIGH PRESS AE900PMP-008	60 m^3/h / 0.3 MPa
Portable transformer AE900XFR001	230/118 V / 3 kVA
Portable transformer AE900XFR002	230/118 V / 3 kVA