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

SLOVENIAN NATIONAL REPORT ON NUCLEAR STRESS TESTS

Progress 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**

Slovenian Nuclear Safety Administration
Železna cesta 16, P. O. Box 5759
1001 Ljubljana, Slovenia
Telephone: +386-1/472 11 00
Fax: +386-1/472 11 99
gp.ursjv@gov.si
http://www.ursjv.gov.si/

URSJV/RP-084/2011

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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 NPP with the net electrical output of 696MWe. 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 in case of beyond design basis accidents.

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. The SNSA made a quick review of the report and already presented some findings and comments to the plant. Some of these have been taken into account in this revision of the report, while others will be addressed in the Final report. The SNSA will proceed with a more detailed review during the next weeks and if needed present additional comments or suggestions to the plant to be resolved in the Final report.

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 progress report would be used as a basis of this national progress report and sent to the European Commission.

Please note that this is not the final report, so changes within the scope and contents, as well as results are to be expected. Additional proposals for improvements can also be expected in the final national report.

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 0.3g was used for Safe Shutdown Earthquake (SSE) and 0.15g 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, hydro-geological, geophysical and engineering seismological investigations. These were performed in several stages. In the seventies the investigations included refractional measurements, soil survey, microseismic 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 0.42g, while the 2004 PSHA study has further increased the seismic hazard to PGA of 0.56g. 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 1 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 2.0g 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 CDF. The new 2004 seismic hazard frequency has increased substantially; a new screening target has been set at 2.75g median capacity with an associated High Confidence of Low Probability of Failure (HCLPF) value of about 1.0g 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 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 0.8g to 0.9g or higher. Seismic events at which early radioactivity releases to the environment would be likely to occur are considered to be at PGA 1.2g or higher and late radioactivity releases in the range of 0.8g to 0.9g. Spent Fuel Pit integrity would not be challenged for PGA's up to approximately 0.9g. For earthquakes exceeding the PGA of 0.9g, gross structural failures of SFP cannot be excluded and fuel uncovers are considered likely to occur.

Seismic events with PGA in the range of 0.8g to 0.9g or higher were estimated to be very rare at the NEK site, the return period is of the order of 100,000 years or larger.

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-Brežiško polje, 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 was the 10,000 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 10,000 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. 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, are considered and for each range the success paths are defined as a minimum set of functions requiring 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 would be loss of offsite power (due to overall conditions in the territory of Slovenia), clogging of intake structures of the essential service water system (ESW) and in an extreme case possible loss of emergency diesel generators. In such cases 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 (CS) 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 black-out (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 400kV grid by one power line towards Maribor, and two towards Zagreb (Croatia). The switchyard 400kV bus is also connected to the 110kV system via 300MVA transformers. For startup and emergency, the Krško Plant is also connected to the 110kV grid by the power line Krško NPP – Thermal power plant 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) 6.3kV buses. If off-site power supply is lost, the two 6.3kV emergency buses MD1 and MD2 are powered from their respective 3.5MW emergency diesel generators. The required power to one train of safety systems is approximately 1.5MW and, with the available fuel at the site, at least 7 days of emergency diesel generator operation for this load is possible.

Each Class 1E train is provided with a complete 125V DC system which supplies DC power to loads associated with the train. Each train’s system consists of a full capacity 125V 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 108V (1.80V 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 2080Ah.

Among the components of the mobile equipment essential for managing severe accidents (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 118V 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 (80 °F) and a minimum of 0.6 °C (33 °F).

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 River Sava. The envisaged alternate way of cooling the loads is by the existing fire protection system with demineralized water available onsite or from the River Sava. 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 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 mobile equipment is stored on safe location for at least first 72 hours. 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 3.4 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 3.4 days.

Possible disruption with regard to the measures for envisaged accidents and associated management is given in the report. 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 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 147.700 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 is a Westinghouse pressurized water reactor with two coolant loops. The Nuclear Steam Supply System and the design and fabrication of the initial cores with 16 x 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 m3 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 380-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 RTP Krško 400kV bus and connected to the 110kV 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 - TE Brestanica via RTP Krško.

The reactor is designed to operate at core power levels up to 1994 MWt, which corresponds to a net electrical output of up to approximately 683 MWe. 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 USAR. The plant is permanently upgraded and modernized in accordance with new industrial and regulatory requirements and standards. The Slovenian Nuclear Law (ZVISJV) 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 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.

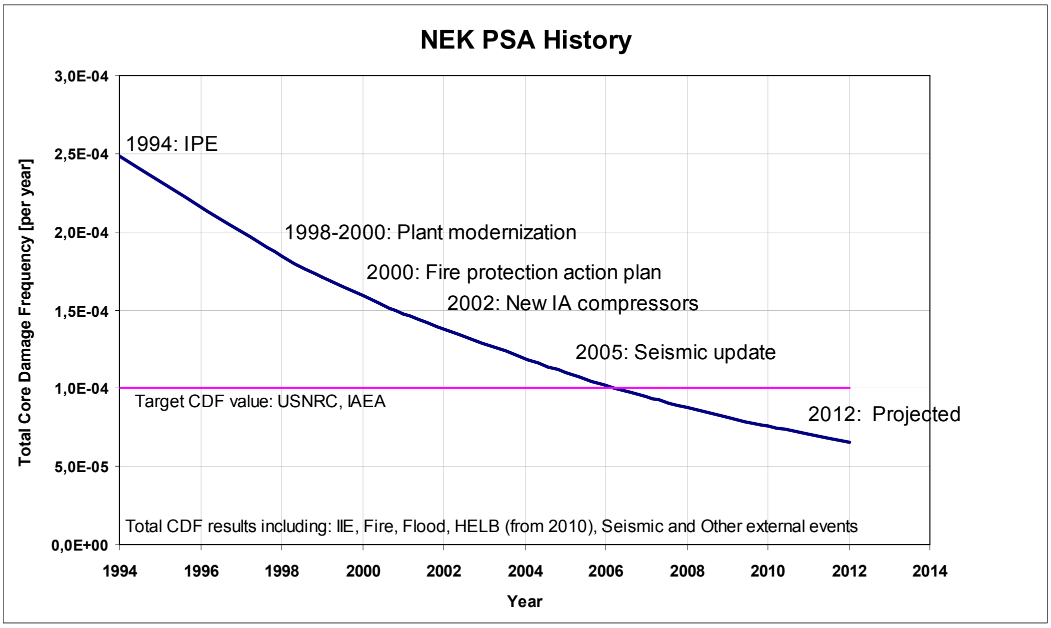

Figure 1: At-power total core damage frequency history [CDF/yr]

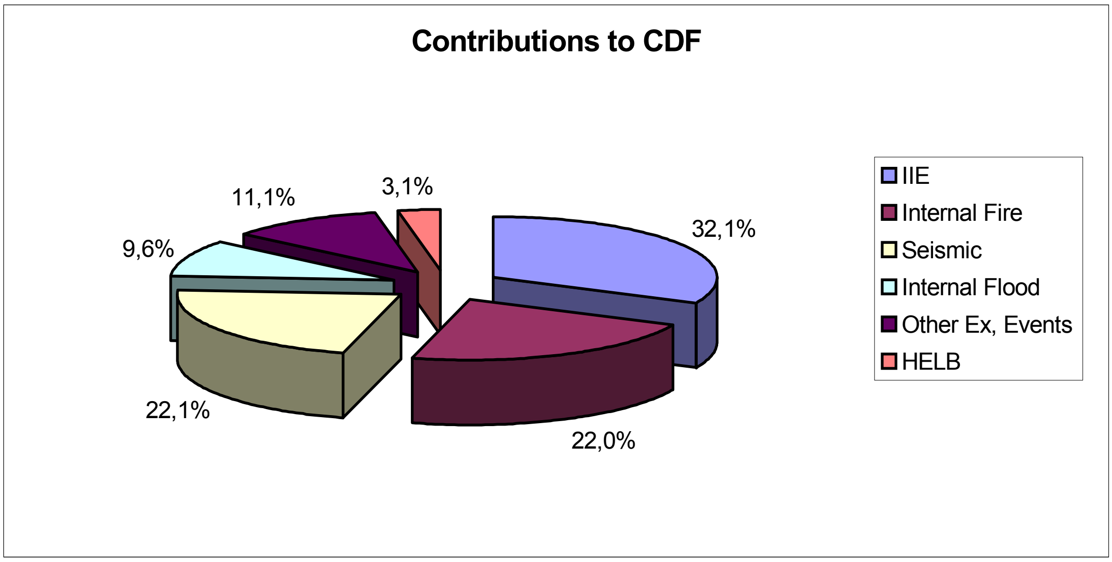

Figure 2: At-power contributions to total core damage frequency [CDF/yr]

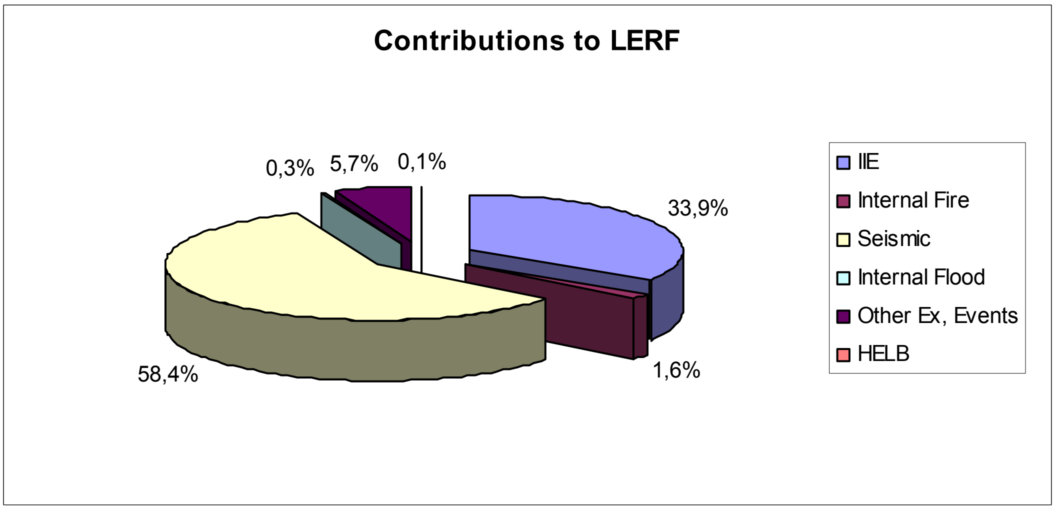

Figure 3: At-power contributions to large early release frequency [LERF/yr]

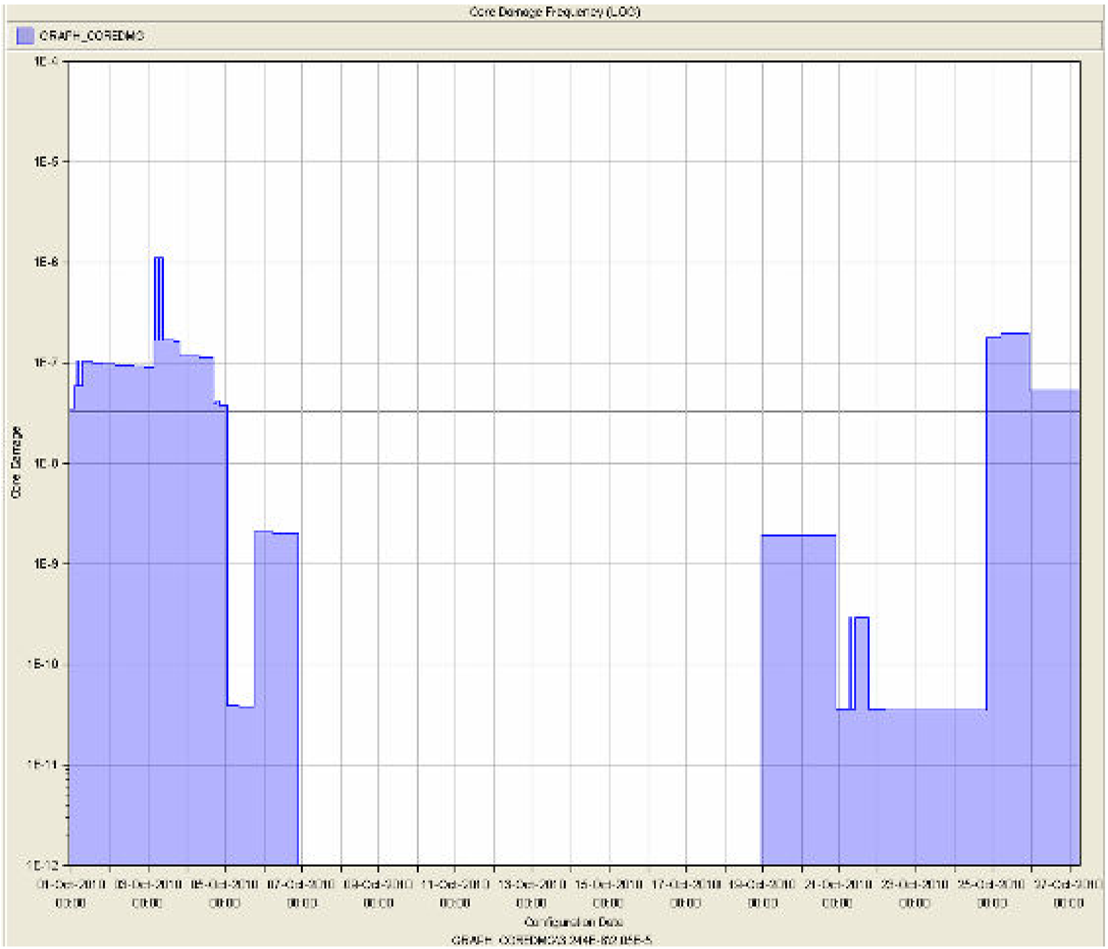

Figure 4: Shutdown core damage frequency profile (Average CDF = 3,24E-08/hr and core damage probability for the time of shutdown CDP = 2,05E-05)

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 (DBE)

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 0.3g was used for Safe Shutdown Earthquake (SSE) and 0.15g 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 are required to operate within design limits. The seismic design for the safe shutdown earthquake 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 behaviour of this structure under the safe shutdown earthquake 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.

2.1.1.2 Methodology used to evaluate the design basis earthquake

The Krško Nuclear Power Plant (NPP) is located along the Sava River in the Krško basin in southeast Slovenia about 2 km from the town of Krško and about 40 km from 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 meters in depth, refractional measurements of P and S wave velocities, geoelectrical trial boring of the terrain, gravimetric soil survey, and microseismic 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 meters in depth, laboratory material tests, and distribution of seismic P and S wave velocities according to cross-hole methodology up to 45 meters 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, gamma-gamma measurements of material density and geoelectrical sounding of the terrain. In addition to the above mentioned investigation, six trial pits to 4 meters 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 (PSA) to evaluate the plant’s vulnerabilities to seismic events. The first step in the seismic PSA, 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 (FGG) 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 1:5,000 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 developed as a part of the first Periodic Safety Review, 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 seismic PSA.

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

As mentioned, the original design peak ground acceleration of the SSE ground motion amounted to PGA=0.3g 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. According to the PSHA study, completed in 1994, a larger peak ground acceleration (PGA = 0.42g) can be expected at the surface. The most recent PSHA (study 2004) of the Krško NPP site has further increased the seismic hazard to PGA = 0.56g. 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 = 0.6g, 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 (SSC) required for achieving safe shutdown state and supposed to remain available after the earthquake

The design criteria used during design of Krško NPP adopted for structures, systems, and components depends 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 Safe Shutdown Earthquake (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 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 structures, systems and components which must be designed to remain functional during the safe shutdown earthquake (SSE) as Seismic Category I. Specifically, if a SSE occurs, all Seismic Category I structures, systems and components must withstand the effects of the SSE and assure:

1. The integrity of the reactor coolant pressure boundary;
2. The capability to shut down the reactor and maintain it in a safe shutdown condition; and

3. The capability to prevent or mitigate the consequences of accidents which could result in potential off-site 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 Pit 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 PIT COOLING SYSTEM

- Heat Exchanger
- Piping
- Pumps

NUCLEAR SAMPLING SYSTEM

- Piping and Valves
- Sample Heat Exchanger

AUXILIARY FEEDWATER SYSTEM

- Auxiliary Feedwater Pumps
    a. Electric Motor Driven
    b. 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

HVAC SYSTEMS

- 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 Pit Area Charcoal Exhaust Fans
- Spent Fuel Pit Area Charcoal Exhaust Plenums
- Spent Fuel Pit 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 40,000m^3. It is designed for a maximum internal pressure of 3.15kg/cm2 rel. with a coincident temperature of 128°C under accident conditions and a maximum external pressure differential of approximately 0.10 kg/cm2 due to accidental operation of the Containment spray System. 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 safe shutdown earthquake (SSE) and Postulated Piping Ruptures are combined using the square root of the sum of the squares (SRSS) method.

Using the principle of seismic design class I Structures, Systems and Components 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 off-site 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 1.0 times the peak 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 200 °F (93.3 °C) or to sizes of 2 inches and under for design temperatures above 200 °F. 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 (RCL), 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 RCL 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 safe shutdown earthquake 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 Pit 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 Updated Safety Analysis Report, 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 1 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 Reactor Coolant System primary components interconnected between floors, the procedures of the following paragraphs are used. The primary components of the Reactor Coolant System 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 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 percent) 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 NE Krško and all design participants who may require it for dynamic analysis or testing of Category I structures, systems or equipment. Coordination of the distribution of this material is performed by the cognizant GAI engineering group.
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 following 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 Seismic PRA 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 CAV (Cumulative Absolute Velocity) 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.

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 (Seismic Probabilistic Seismic Analyses), 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 6300 V safety bus:
  a. The faulted 6300 volt bus will be isolated. The corresponding 400 volt 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.
  b. 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.
  c. The faulted bus will be isolated by protective circuit breaker action so that minimal 400 volt auxiliaries will be lost.
  d. 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 6300 V 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 system 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. NE Krško 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 seconds and re-energization 0.3 seconds after clearing. Actual line protection will enable fault clearing within 0.1 to 0.15 seconds.

After consideration of the likelihood of losing off-site 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 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 2.000 meters 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 southeast of the site, has a 5.000 meters 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 northwest of the site, has a 3.000 meters 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 military airport Cerklje which is also used for civil purposes and is only 10 km away from the plant.

2.1.2.5.4 Fire and explosion

The Fire Protection System is designed to provide adequate fire protection from all known 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 three hour fire walls. These buildings are: reactor building, auxiliary building, control building, fuel handling building, intermediate building, diesel generator building, turbine building,

žghand 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. 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 Fire Protection System is designed in accordance with the intent of the requirements for fire protection for nuclear plants established in the following:

1. The National Fire Protection Association (NFPA) - National Fire Codes.
2. 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 United States manufactured fire protection equipment is approved for fire protection use by Underwriters' Laboratories Inc. (UL) or Factory Mutual Engineering Corp. (FM). The Fire Protection System is designed such that the operation or failure of any portions of the Fire Protection System 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 fire protection system 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 fire protection 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. Fire protection piping within the seismically qualified charcoal filters and near the reactor coolant pumps are supported to withstand a seismic event. All fire protection 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 Fire Protection System has been performed showing that failure of any portion of the Fire Protection system would cause "Fire Alarm" and "Trouble Alarm" annunciated at the Main Control Room 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 Nuclear Power Plant is presented in such a way, that all personnel are performing quality-affecting activities on-site. 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 (MD), Technical Operation Manual (TD), Engineering Services Manual (ED), Quality Systems Manual (QD), Purchasing Manual (PD), Financial Manual (FD), Information System Manual (IS), Security Manual (SD), Training and Indoctrination Manual (TI).

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 (QAP), 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 ED 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 (KOC), 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 (KOC) and prepared as a basis for the review by Krško Safety Committee (KSC), to verify that the modification is properly evaluated per 10CFR50.59.

<u>Procurement and document control</u>

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 suppliers 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 (DCM) 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.

<u>Regulatory requirements</u>

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 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 Compliance Program ("RCP") which represents the overview of all US NRC regulatory requirements and NPP compliance with them. This document is also a part of the Periodic Safety Review process. Currently, revision 3 is in place.

<u>Conclusion</u>

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.

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). 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 is performed once per year). The complete list of SAME mobile equipment is provided in Table 6 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, …) are 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 (JTA) 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 Probabilistic Safety Hazard Analyses (PSHA) analyses in 1994. The resulting hazard determined within this study was used as an input for seismic PSA 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 (UHS) corresponding to a return period of interest. For the NPP, the UHS shape corresponding to a 10,000 year return period was used. This UHS 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 0.3g and median UHS for the probabilistic analyses were anchored to a peak ground acceleration (PGA) of 0.6g.

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 45km seismic profiles.

As a part of the seismic PSA investigations for the NPP, IPEEE analysis for the seismic part was performed in the nineties. An Individual Plant Evaluation (IPE) was performed for the NPP including the Individual Plant Examination for External Events (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 SPSA (1996). Also, a 0.6 kV portable diesel generator at the site was examined for purposes of gathering information for development of a 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 0.6kV 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 MCCD 221.

In December, 2003 a more detailed walkdown was conducted outside of containment of all components noted in the 1994 walkdown report (1994) to require upgrading or additional assessment or to assess new equipment added or replaced since the 1996 SPSA.

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 SPSA. 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 (RWMUT). 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 2,75g median PGA screening capacity. In the 1995 SPSA the screened out level was above 2.0g. In many cases the calculated fragility was much higher and the accumulated conservatism of only crediting the SSCs with 2.0g median capacity tended to overestimate the CDF. The screening level has been raised to 2.75g which will result in a lower probability of failure when convolved with the new hazard than the previous 2.0g fragility when convolved with the 1994 hazard.

The new seismic hazard screening level raised from 2.0g to the higher level of 2.75g 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 Component Cooling Water Surge Tank, High Pressure Safety Injection 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 1996 SPSA 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
- GH compressor unit improvement

- Polar crane seismic improvement
- eismic protection of polar crane
- Installation of seismic stabilizer
- Improvement of DG fuel tank supports
- Seismic reinforcement of IA tanks
- Improvement of nitrogen bottles attachment

The first Periodic Safety Review 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 majority of required actions required by US NRC (NEI 06-12 B.5.b).

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 Safety Analysis Report 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 SSC (Systems, Structures and Components).
- 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 HCLPF (High Confidence of Low Probability of Failure) 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 a peak ground acceleration (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 Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) levels were defined in the FSAR 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 Individual Plant Examination (IPE) was undertaken, including internal and external events and hazards. As a part of it, a probabilistic seismic hazard assessment (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 probabilistic seismic response analysis and seismic probabilistic risk analysis (PRA).

In the probabilistic seismic response analysis, which was conducted under the IPE for External Events (IPEEE), a detailed assessment was performed of floor response spectra, based on the input hazard spectra anchored to 2 x 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 seismic PRA.

Seismic Probabilistic Risk Analysis 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 an NPP.

In the early 2000s, the PSHA for NPP Krško site was revisited as a part of the first Periodic Safety Review (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 Seismic PRA for Krško NPP (including PRA Level 1 and Level 2), which was performed, roughly, a decade later than the initial seismic PRA 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 Nuclear Steam Supply System 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 LOCA, the required critical function for success (i.e. avoidance of core damage) would be Low Pressure Safety Injection and Recirculation.

If induced initiator is Medium LOCA, the required critical functions for the success would be Reactor Scram and High Pressure Safety Injection / 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 that induced initiator is Small LOCA, the required critical functions for the success would be Reactor Scram, High Pressure Safety Injection / 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, the required critical functions would be Reactor Scram, High Pressure Safety Injection / Recirculation and Secondary Heat Sink (by means of AF MDPs). (Note: There are also some other success paths involving successful MS 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 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 CS Charging Pumps).

If none of the above initiators is induced, the Loss of Offsite Power should be considered. In such a case, the required critical functions would be Onsite Power (by means of Emergency DGs), Secondary Heat Sink (AFW) and RCP Seal Injection (by means of CS 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, …).

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 a peak ground acceleration (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 the OBE value, failure of the 380 kV switchyard failures cannot be excluded. However, a failure of 110 kV source is considered low probability event at this seismicity level. Therefore, a complete Loss of Offsite Power is not considered likely. Failures of any safety related SSC are considered unlikely. 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, 380 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 Loss of Offsite Power (LOOP) can be expected. Failures of safety related SSC are considered unlikely. 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 110 kV offsite power available. None of the critical functions required for any of the success paths considered is affected in this range.

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 a LOOP without additional failures of safety related SSCs. None of the critical functions required for any of the discussed success paths is considered to be affected.

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 CST and / or RWST. 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 CS Pumps taking suction from the 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 CS Pumps suction from the VCT). However, considering the seismic capacity of CS 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 possible. Failure of RWST is not considered a concern, as far as induced LOCAs /SLB are not considered, due to the high seismic capacities of underlying SSCs (as discussed above). The RWST is also alternative source for CS Pumps to perform RCP Seal Injection. This function (suction to CS Pumps) can, however, be achieved by alternative means described in the EOP ECA-0.0 Appendices. Regarding the other required functions, seismically induced Reactor Scram Failure (based on the impact on Fuel Assembly Geometry) and Loss of ESW (due to ESW Pump House seismic failure modes) are not considered likely in this interval. Similarly, failure of DGs is not considered likely in this range. These considerations are based on the corresponding seismic capacities. Assuming the failure of CST, alternative suction would be established for the AFW Pumps by re-alignment to ESW. Assuming the additional failure of DGs, although it is not likely in this range, alternative means would be required to support the success paths considered, as DG availability is necessary for all success paths discussed earlier. Alternative means, as described in the EOP ECA-0.0 Appendices, will be needed to ensure secondary heat sink and RCP Seals integrity. The first would be implemented by providing cooling water to SGs directly from alternative source or by AFW TDP from alternative suction. (The AFW TDP flow would be controlled manually.) The second would be provided by the CS PDP powered from a 400 V Diesel Generator connected to the LD bus.

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

At this interval, seismic failure of EDGs is considered likely. This means that alternative means would be needed to ensure secondary heat sink and the RCP Seals integrity. Those were discussed above. Additionally, at the upper part of the interval, failures of Control Rods Insertion and loss of ESW Pump House would become likely. Loss of ESW would not have additional impact, as DGs would, likely, be also lost at this level. Therefore, critical functions would be those associated with seismic ATWS sequence. Long Term Shutdown (sub-criticality) is not considered to be a concern due to high seismic capacity of BA Transfer Pumps / Tank and CS PDP. RCS inventory would also be ensured by CS PDP with suction aligned to alternative means (ECA-0.0 Appendices). Therefore, the critical function would be ensuring the secondary heat sink following a seismic ATWS. It is questionable whether, in the ATWS sequence, the alternative suction to AFW Pumps can be ensured in time. To summarize: at the lower end of the range, the expected sequence is seismic SBO with structural failure of CST and RWST; toward the upper end, the expected sequence is seismic ATWS with SBO conditions (which encompass the loss of ESW).). The point where seismic core damage would become likely is assessed to be at 0.8 g – 0.9 g.

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, DGs 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 above evaluation, seismic levels at which core damage would be likely are considered to be at the PGA grange of 0.8 g to 0.9g 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. 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 those seismic events with PGA in the range of 0.8 g to 0.9 g (or higher), at which reactor core damage is considered likely, were estimated to be very rare events at NEK site. Based on the plant specific studies, the return period for such an event is of the order of 100 000 years or larger.

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 CS 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

This section addresses seismically induced containment failure and not containment failure which might happen in the sequence of a severe accident such as induced by core damage in a seismic event with a PGA greater than 0.8 – 0.9 g.

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.
    iii. 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 (Systems, Structures and Components).
- 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 a peak ground acceleration (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 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 BDB 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.

<u>Evaluation of the seismic margin for containment integrity</u>

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 a peak ground acceleration (PGA).

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

<u>*Earthquakes in the range of PGA < 0.45 g).*</u>

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.

<u>*Earthquakes in the range of 0.45 g < PGA < 0.60 g).*</u>

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. 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 Containment Spray / 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. 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. 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. 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 CS 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. 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 – 0.9 g. 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. Those strategies would ensure slow containment heatup 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, DGs 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 at some 1.2 g or higher. 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 strategies, noted above (timing, scrubbing, planning of the releases).

<u>Conclusion regarding the seismic margin for containment integrity – cliff edge effect</u>

Early releases

Seismic events at which early radioactivity releases into the environment would be likely to occur are considered to be of PGA as high as 1.2 g or higher. 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 to 0.9 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. Those strategies would ensure that any release to the environment is limited. The releases are expected to be scrubbed and planned.

<u>Spent Fuel Pit</u>

For Spent Fuel Pit, 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 Seismic PRA. 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 spent fuel pool 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 (0.14 g for BWR and 0.2 g for PWR).

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 NPP SFP, as NPP was designed and constructed in accordance with design requirements for US plants. Therefore, seismic capacity regarding the gross structural failure of SFP is considered to exceed 0.9 g. 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 Loss of Offsite Power 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 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 FA 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 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 is considered likely.

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 are credited to provide the makeup water for the SFP inventory and, thus, prevent the FAs 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 is considered likely that fuel uncovery in the SFP would occur.

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

2.2.3.1 Physical impact

For the consideration of seismically induced floods, relevant are hydro power plant 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. Hydro power plants 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 hydro power plants 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 hydro power plants (HPP) at upper Sava River is comprised of HPP Moste, HPP Mavčiče and HPP Medvode. They are all operating plants.

The group of hydro power plants at lower Sava River is comprised of:

- HPP Vrhovo, which is an operating plant;
- HPP Boštanj, which is under construction;
- HPP Blanca, HPP Krško, 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 has 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 m3/s and duration of 8 hr. 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 9th or 10th grade on the EMS scale, or an earthquake with the magnitude well above 6.0. In terms of PGA, this would mean an earthquake in the range of 0.6 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.

2.2.3.2 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 indentified.

2.2.3.3 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 plant is designed

Protection from floods was accomplished by the plant design and construction of the Sava river dikes upstream and downstream the plant.

Plant Building entrances and openings are constructed above the elevation of the 10,000-year flood. So the plant is, even without protection dike, safe for the occurrence of the design flood.

Beside 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.

Against local heavy rainstorm the plant is protected with the plant design and drainage system.

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 m3/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, located in the centre of the plant site, as can be observed on the Figure 5, 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 favourable 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 m3/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 (note that plant is located on the left Sava river bank, refer to the Figure 5). The design flood elevation of the interfacing structures is 156.50 m.a.A.s.l., considering 0.6m of safety margin the elevation of the structures plateau is located at 157.10 m.a.A.s.l.

Figure 5: Krško NPP site location

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. These functions can be fully defined if the numerical values of their parameters are known. Theoretical probability distributions to be selected from an analysis are generally limited to those that can be defined by the three following parameters: arithmetic mean, standard deviation and coefficient of asymmetry.

Determination of the distribution function that will in the best possible way describe it, is characterized by the following steps:

a. determination of numerical values of the parameters of distribution function
b. elaboration of accuracy parameters
c. selection of adequate distribution function

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.

a. Defining parameters of distribution functions

There is an extensive data base of the Sava river water level and flow measurements available. According to the data obtained, observations spanning over 74 years are available for the period from 1926 – 2000.

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		

Also available was flow recorded around 36 km upstream of Krško in Radeče.

Table 2: Maximum Sava river flow in Radeče [m3/s]; Period: 1908 – 2000

Year	Flow m^3/s	Year	Flow m^3/s	Year	Flow m^3/s	Year	Flow m^3/s
1908	1317	*1931*	1661	*1954*	1536	*1977*	1370
1909	1863	*1932*	1325	*1955*	1068	*1978*	1216
1910	1119	*1933*	2809	*1956*	1420	*1979*	2498
1911	1131	*1934*	1998	*1957*	1000	*1980*	2391
1912	1104	*1935*	1536	*1958*	1309	*1981*	942
1913	1007	*1936*	2228	*1959*	1420	*1982*	2313
1914	1321	*1937*	1380	*1960*	1458	*1983*	1450
1915	1754	*1938*	1482	*1961*	2024	*1984*	1416
1916	1466	*1939*	1458	*1962*	1789	*1985*	1831
1917	1513	*1940*	1772	*1963*	1661	*1986*	1492
1918	1458	*1941*	849	*1964*	2699	*1987*	1857
1919	1400	*1942*	805	*1965*	1957	*1988*	1331
1920	832	*1943*	1341	*1966*	2350	*1989*	1391
1921	1420	*1944*	1264	*1967*	1712	*1990*	2991
1922	1576	*1945*	815	*1968*	1708	*1991*	1862
1923	2676	*1946*	582	*1969*	1839	*1992*	2153
1924	997	*1947*	1412	*1970*	1346	*1993*	1735
1925	1576	*1948*	1693	*1971*	973	*1994*	1217
1926	2109	*1949*	1820	*1972*	1913	*1995*	1390
1927	2134	*1950*	929	*1973*	2460	*1996*	1865
1928	1294	*1951*	1552	*1974*	2213	*1997*	1250
1929	805	*1952*	1302	*1975*	1930	*1998*	2940
1930	2087	*1953*	1018	*1976*	1380	*1999*	1133
						2000	2080

There are several methods for estimating statistical population parameters from available data base. For the first five functions of distribution, mainly the method of instant is used, while for the sixth function parameters by the maximum probability method are defined.

Result gained from statistical distributions is flow with return period of 10000 years.

Table 3: Distribution of flood flows

Distribution Law	Return period (year)	2	5	10	20	50	100	1000	10000
	Probability (%)	50	20	10	5	2	1	0.1	0.01
	Data source	**RETURN PERIOD FLOW (m^3/s)**							
Log Pearson Type III	Krško 1926-2000	1630	2161	2550	2952	3517	3978	5775	8100
	Radeče 1908-2000	1498	1982	2311	2632	3059	3390	4564	5894
	Radeče 1951-2000	1619	2081	2384	2675	3053	3340	4328	5401
	Krško 1951-2000	1743	2200	2497	2778	3142	3416	4347	5340
Gumbel's Law	Krško 1926-2000	1675	2131	2433	2723	3098	3379	4308	5236
	Krško 1951-2000	1740	2172	2458	2733	3088	3355	4235	5113
	Radeče 1951-2000	1620	2068	2364	2648	3016	3292	4203	5113
	Radeče 1908-2000	1511	1961	2260	2546	2916	3194	4111	5027
Log Normal (2P)	Krško 1926-2000	1688	2149	2440	2710	3049	3295	4048	4695
	Radeče 1908-2000	1519	1973	2265	2539	2885	3138	3925	4609
	Radeče 1951-2000	1632	2084	2370	2637	2971	3214	3959	4599
	Krško 1951-2000	1758	2194	2465	2715	3026	3249	3926	4498
Pearson Type III	Radeče 1951-2000	1643	2102	2379	2627	2927	3140	3795	4398
	Krško 1926-2000	1709	2173	2445	2686	2975	3178	3793	4351
	Radeče 1908-2000	1539	1999	2273	2517	2811	3019	3653	4234
	Krško 1951-2000	1776	2214	2468	2693	2961	3148	3714	4225
Gamma (2P)	Krško 1926-2000	1707	2182	2462	2710	3008	3217	3853	4431
	Krško 1951-2000	1778	2208	2458	2678	2940	3123	3675	4173
	Radeče 1908-2000	1542	1995	2264	2502	2790	2992	3609	4171
	Radeče 1951-2000	1656	2095	2352	2579	2851	3042	3620	4143
Normal (2P)	Krško 1926-2000	1759	2194	2421	2609	2820	2961	3355	3680
	Krško 1951-2000	1820	2232	2447	2625	2825	2958	3332	3639
	Radeče 1951-2000	1703	2129	2352	2536	2743	2882	3268	3587
	Radeče 1908-2000	1594	2023	2248	2433	2642	2781	3170	3491

b. Analysis of distribution function accuracy

The elaboration of accuracy for above statistical parameters, estimated from samples, is based on two methods:

- Chi - Square Goodness of Fit Test. The Goodness of Fit (GoF) tests are statistical procedures to establish whether an assumed distribution is correct and how measured data follow this distribution. If the distribution is correct, its Probability Density Function (yielding an area of unity) should closely encompass the data range (of X). If data do not support the distribution, it should be rejected.
- Kolmogorov – Smirnov test adjustment probability.

c. Selection of adequate distribution function

If we consider Chi - Square Goodness of Fit Test criteria (statistical conclusion) 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 1926 - 2000 is 4431 m3/s.

To be on the safe side we may choose Log Normal distribution. In this case the 10000 – year flow of the Sava river at Krško is about 4700 m3/s.

Table 4: Long return period floods and χ^2 values

Distribution Law	Return period (year) / Probability (%)	100 / 1	1000 / 0.1	10000 / 0.01	Criterion $\chi^2 < 5.99$
Gamma	**Krško 1926-2000**	**3217**	**3853**	**<u>4431</u>**	1.78
Pearson III	Krško 1926-2000	3178	3793	4351	2.41
Log Normal	Krško 1926-2000	3295	4048	**4695**	14.49
Gamma	Radeče 1908-2000	2992	3609	4171	0.695
Pearson III	Radeče 1908-2000	3019	3653	4234	0.5
Log Normal	Radeče 1908-2000	3138	3925	4609	2.28

According to Kolmogorov – Smirnov adjustment tests presented, in order to define 10000 – year flow, we can say that all six functions could be applied. According to Kolmogorov – Smirnov test Pearson III (Q10000 = 4351 m3/s) and Gamma (Q10000 = 4431 m3/s) functions are most qualitative ones. So it can be said that the most probable 10000 – year flow of the Sava river at Krško is about 4431 m3/s. To be on the safe side the Log Normal function can be selected and as such the 10000-year flood flow of the Sava river at Krško is about 4700 m3/s.

Table 5: Kolmogorov – Smirnov fitness test; Data: Krško 1926 – 2000

Distribution	Normal	Pearson III	Gamma	Log Normal	Gumbel	Log Pearson III
Year of d_{max} value	1959	1931	1931	1931	1931	1970
Flow at d_{max} value (m^3/s)	2009	1735	1735	1735	1735	1683
Observed / empirical probability P_o	0.737	0.447	0.447	0.447	0.447	0.408
Theoretical probability P_t	0.685	0.520	0.522	0.538	0.551	0.541
value d_{max} = abs(P_o-P_t)	**0.051**	**0.073**	**0.074**	**0.091**	**0.103**	**0.133**
Variable $Z = d_{max} \cdot (75)^{0.5}$	0.4452	0.6303	0.6424	0.7850	0.8944	1.1507
Probability of adjustment	**98.9%**	**82.2%**	**80.4%**	**56.9%**	**40.1%**	**14.2%**
10000 – year flow (m^3/s)	3680	4351	4431	4695	5236	8100

As the national value of Institute for Water of the Republic of Slovenia, 10000 – year flood calculated with Log Pearson III distribution is 4790 m3/s, this flow is selected as conservative value of 10000 – year design flood.

According to adjustment tests in order to define 10000 – year flow, it was concluded, that all six distribution functions could be applied for 10000 – year flow determination. However even if the Normal distribution contains the highest adjustment probability, P = 98.9%, we know that this distribution gives too small 10000 – year flows. Likewise Log Pearson III (CDF Integral method) and Gumbel distributions were rejected because they give too high flow values and also according to Kolmogorov – Smirnov test their adjustment probabilities are poor (only 14% and 40%). According to Kolmogorov – Smirnov test Pearson III (P = 82%, Q10000 = 4351 m3/s) and Gamma (P = 80%, Q10000 = 4431 m3/s) functions are most qualitative ones. To be on the safe side the Log Normal function can be selected and as such the 10000-year flood flow.

Probable maximum flood

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

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

A brief description of each step is provided in the following subchapters.

a. Determination of probable maximum precipitation (PMP)

According to the definition given by the "American Meteorologic Society", 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". This value can be determined if we combine theoretical and empirical approaches by means of air moisture content or by transposition of atmospheric disturbance. Often enough, instead of probable maximum precipitation, the so called standard storms are taken as reasonably strongest acceptable atmospheric disturbance, which characterizes pluviometrically a certain region. A.K.Biswas suggests that the highest recorded amounts of precipitation be increased by 100%, in order to obtain the strongest atmospheric disturbance. Consequently, this would mean probable maximum precipitation on the drainage area of 7850 km2.

b. Determination of runoff loses with runoff model

In order to estimate the probable maximum flood flows from probable maximum precipitation, it is necessary to study thoroughly the runoff losses. Between precipitation and runoff there are many factors that have effects on this relationship, thus creating the runoff deficiency in the form of infiltration, evaporation, evapotranspiration.

In the previous Chapter we have defined the probable maximum effective precipitation on the Sava drainage basin downstream Krško as 365.1 mm, which means that we are dealing with the volume of water amounting to 0.3651 x 7850 x 106 = 2866 million m3, which will cause the flood wave. In the further computational procedure it is necessary to shape the flood wave analytically.

c. Determination of probable maximum flow and level

By choosing the appropriate shape of flood wave from the history of floods, the volume of flood wave of 2866 million m3 was calculated to produce the 6500 m3/s flood flow. This flow covers PMP.

Probable maximum flow is 6500 m3/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 river bank inundation.

d. Casual waves' activity due to wind

Analysis of wind activity according to Stevenson has been carried out. Under supposition that wind velocity is 21 m/sec 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. Consequently, it is possible to expect raising of water surface at the nuclear power plant for 46 cm, respectively at the elevation 156.16 m.a.A.s.l.

e. Potential danger due to upstream dam's failure (seismic origin)

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

Evaluation of flood wave propagation has been carried out, under supposition that breaking of dam 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 under such supposition, the flood wave of 25 year floods in Krško would rise to the value of 3333 m3/s, which approximately corresponds to 1000 year flood. Consequently, the safety of nuclear power plant facilities in Krško would not be endangered by the breaching of the existing hydropower plant structures.

3.1.1.3 Conclusion on the adequacy of the design basis for flooding

As discussed in previous chapters, Krško NPP differentiates two protection levels regarding to flooding.

Plant Building entrances and openings are constructed above the elevation of the 10000-year flood. So the plant is, even without protection dike, safe for the occurrence of the design flood.

Beside 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.

The design with additional protection dike is therefore adequate for flooding. 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.

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)

Not applicable.

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 and to line the generated power.

The switchyard and power lines are located on the left Sava river side, protected in case of flooding. There is relatively small area affected in the case of flooding, considering area locally around the power plant likewise nationally, as Slovenia is hilly country and thus flooding affects only the valley areas. So, there is a very small probability that external power would become unavailable due to flooding.

However, if the probable maximum flood would occur it would come from probable maximum precipitation and in such a case there is a reasonable possibility that nationwide power availability problem would occur due to extensive raining, other area flooding and landslides that might occur. 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 110kV dedicated power line over the hill to the nearest Brestanica thermo power plant and the switchyard placed on the left river side with small possibility to become unavailable even in the case of probable maximum flood. The Brestanica power plant has a procedure to deliver the power solely to Krško NPP in cases of nation wide power loss.

In case of offsite power unavailability, 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 absolutely safe by the occurrence of design flood and they are protected with the dike in case of probable maximum flood. The cooling of the diesel generator engine is independent from Sava river.

3.1.2.2 Main associated design/construction provisions

Main provision against flooding 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 finished grade, 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 means of design of interface structures and by protection dike.

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. Design flood of Sava river flow is 4790 m3/s and corresponds to the 155.35 m.a.A.s.l. Probable maximum flood flow is 6500 m3/s and corresponds to the 155.89 m.a.A.s.l. water level at the dike.

At flow 700 m3/s increased monitoring of Sava river flow, elevation, pollution and cooling systems begins.

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.

As already described, 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.

3.1.2.4 Other effects of the flooding taken into account

3.1.2.4.1 Loss of external power supply

In case of offsite power unavailability, 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 absolutely safe by the occurrence of design flood and they are protected with the dike in case of probable maximum flood.

For complete explanation refer to 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 for power cooling and separately for safety cooling.

The Essential Service Water System (safety and component cooling) 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. The system basically consists of two independent loops. One safety class, automatic, self-cleaning strainer is installed for 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 trash rack with a movable trash rake followed by a travelling water screen. The two channels are connected through two 0.91m (36 inch) openings located on the inlet side of the trash racks and discharge to a common fore bay serving the three pumps. The inlets, channels and travelling 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.

The Circulating Water System provides cooling water for the main condenser and turbine building auxiliary coolers, 25 m3/sec 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 travelling water screens, a travelling water trash rack, screen wash pumps, strainers, and auxiliary service equipment.

Additionally to the trash rakes and travelling screen filters, procedural steps are taken in case of expected or unexpected pollution. 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, is relatively short and top 90% of longest flooding flow lasted around 17 hours. As the plant is not directly jeopardized by the probable maximum flood, the existing personnel at the plant could operate the plant.

However, by the occurrence of the increased flow of 3765 m3/s, the plant alert is initiated. In such a case, technical support centre and operation support centre are called to the plant site. If flow would still increase, at 4274 m3/s, site emergency would be initiated and offsite emergency operations facility would be called. At this flow all roads are still available.

In situations, when access to the plant site would be unavailable (at flows higher than PMF), offsite alternative location for technical support centre and operation support centre would be initiated. The intervention stuff has telephone and radio communication available at the flooding safe 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 Plant 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 rework

The original design flow was 4272 m3/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 (due to Krško NPP Periodic Safety Review) is 4790 m3/s and corresponds to the 155.35 m.a.A.s.l. (according to revised flow level relationship tables) water level at the dike and provides the corresponding safety degree for plant structures (auxiliary building and DG building openings are at the altitude of 155.50 m.a.A.s.l.).

Revised probable maximum flood flow remained 6500 m3/s and corresponds to the 155.89 m.a.A.s.l. water level at the dike.

PMF level was revised, the dikes around the plain exceed this level by 118 - 138 cm. Outside the reach of this plain, the crest elevation of the dike is 98 cm above PMF level. It should be mentioned, that under present conditions on the right river side the level difference between the crest of the dike and floods is favorable (water level at the dam is 155.89 by the occurrence of PMF and the level of dike is 157.10 m.a.A.s.l.). 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.

Comparing the original data and revised results, the difference in the level results may be observed (flood flow increase from 4272 m3/s to 4790 m3/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, even though the flow increased by, 518 m3/s may be found in the assumptions, valid 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 x 20 m resolution for this area. Computer software tool PCFLOW2D was used for dynamic water flow calculations. For the sake of protection of nuclear power plant it was essential that the degree of flood protection on the right side of the Sava river, as well as the elevation of these dikes, was estimated reasonably. It is quite clear that on the occurrence of the probable maximum flow, the Sava river would inundate and the flood flow would be directed towards the areas behind the right river bank. Consequently, it was considered that the right Sava dike should be 1 m lower than the left dike, or the cassette (polder). This minimum request should be taken as postulate for any Sava river bed regulation.

3.1.3.4 Specific compliance check already initiated by the operator

A number of studies and analyses related to external flooding hazard has been performed for NEK during the past decade (refer to chapter 3.2.2).

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 and the NPP openings are at the level of 155.50 m.a.A.s.l. and NPP plateau is at elevation of 155.20 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 6 the situation at PMF flow of 7081 m3/s is presented with current status of protection dikes.

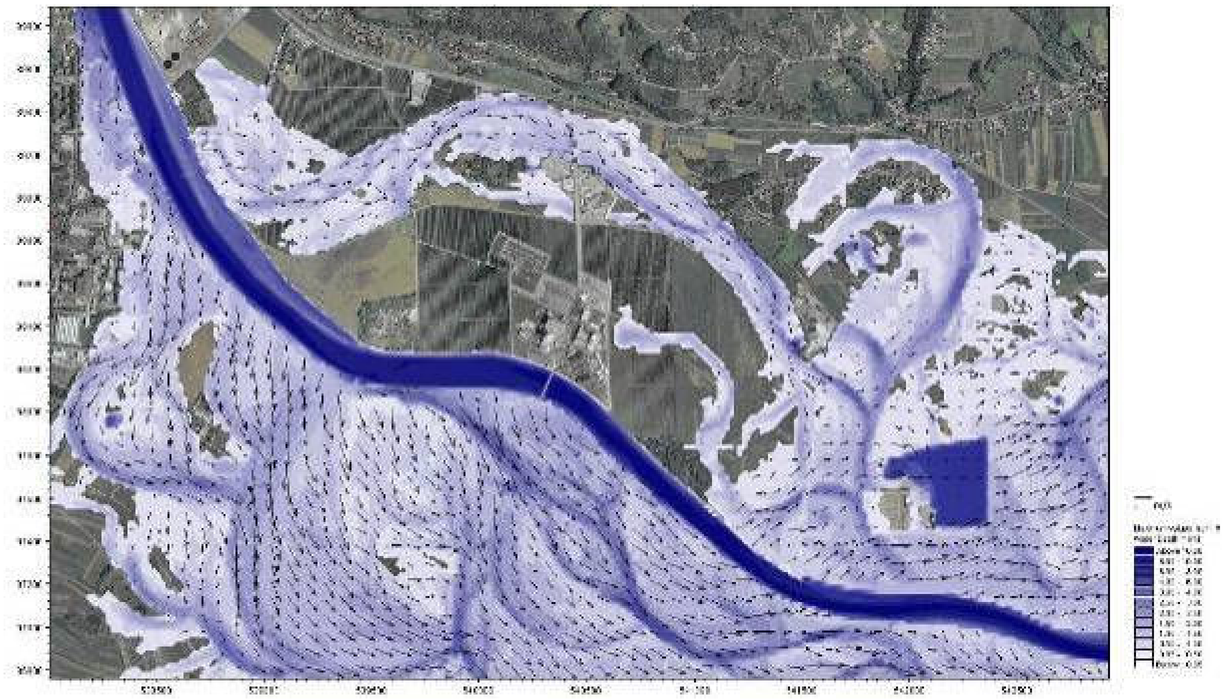

Figure 6: Situation at PMF flow of 7,081 m3/s is presented with current status of protection dikes

After the completion of modification (increasing the dikes upstream of the plant), the flood at the PMF flow of 7,081 m3/s is presented in Figure 7.

Figure 7: After the completion of modification (increasing the dikes upstream of the plant), the flood at the PMF flow of 7,081 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 10,000m3/s is shown on the Figure 8.

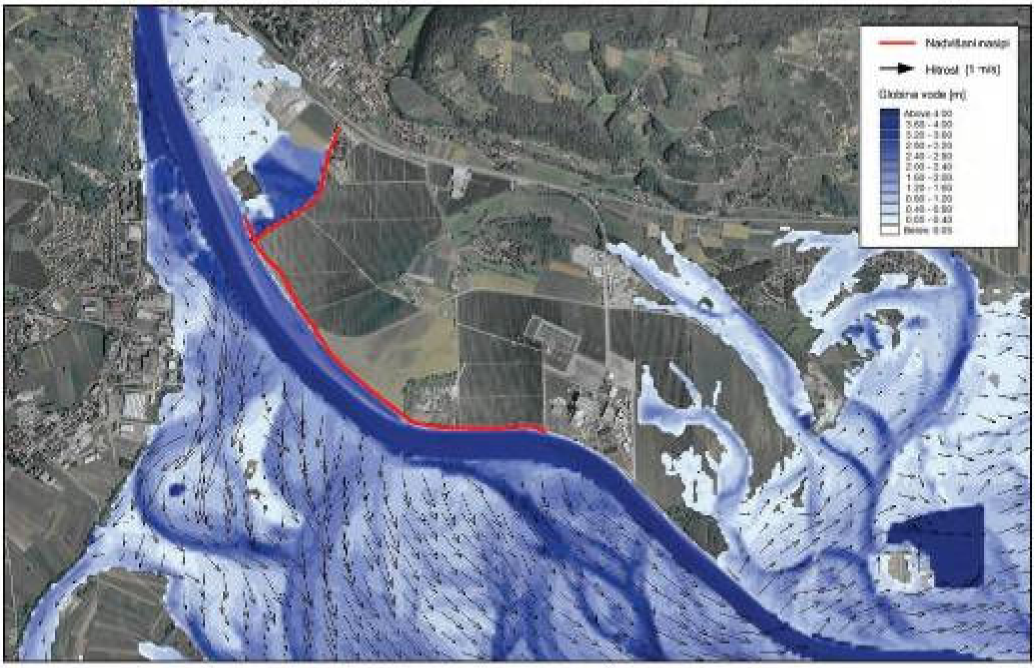

Figure 8: Increased level of dikes upstream of the plant will be capable of protecting the plant to the flood flows beyond the PMF flood. The flood flow of 10,000m3/s

3.2 Evaluation of margins with respect to flooding

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

From the plant specific studies concerning the flooding hazard, two points can be made:

- The flooding event would be relatively slow in progression: The shortest time, considering various scenarios analyzed, for developing the maximum flow would be at or above some 20 hr and, depending on the particular scenario, there could be a considerable delay;
- Judging from the shape of this wave and the waves from the later studies, the duration of the high flows would not exceed the period of one day (24 hours). This however should not be mixed with the duration of high water levels in the flooded area. The discharge time can be expected to be considerably longer.

In the light of the above, the protective measures which can be envisaged based on the warning time would include:

- Implementing normal plant shutdown;
- 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

Approach

Similar approach to evaluation of external flooding margins at NPP is taken to the one used for the evaluation of seismic margins. When applied to the external flooding hazard, it consists, in general, from the following main steps:

- Use the USAR and available safety studies and analyses to:
  1. Establish the relations between relevant elevations at NPP site and flooding water levels achieved at different river flows.
  2. Identify the “success paths” for a range of flooding events. As before, 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.
- Map each critical safety function in every success path to the specific plant’s SSC (Systems, Structures and Components).
- Determine the location of each relevant SSC (e.g. building, elevation).
- Evaluate the availability of all success paths following a postulated flooding event, with increasing severity. Start with the lowest flooding events 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.

Brief history overview

The design requirements for NPP regarding the flooding from Sava River were defined in the FSAR, based on the information on the past registered floods, high waters and applicable requirements regarding the NPP design. Those are based on the flood (river flow) with 10000 year return period. The FSAR also provided the estimate of Probable Maximum Flood (PMF).

During the history of NPP operation, up to now, 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-yr flood, based on the updated history of maximum river flows, and confirmed the adequacy of NPP design requirements regarding the 10000 yr return period. The study also included the re-evaluation of the PMF, which ended with conclusion that the existing PMF estimate in Updated FSAR is still applicable.

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 NPP Krško.

Accompanying this study was another one, representing hydraulic analysis of high Sava River levels at NPP site for the revised Probable Maximum Flood (PMF).

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.

<u>Relevant elevations at NPP site and flooding water levels achieved 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-yr flow is 155.35. (All elevations and levels considered in this and other sections related to the external flooding hazard are expressed in "meters above Adriatic Sea level" (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 on155.50.

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

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. The Essential Service Water and Circulating Water Buildings are at the same elevation. Considering the NPP plain elevation of 155.50, the difference between the dike and the openings of buildings located on the plain is 157.10 – 155.50 = 1.60.

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-yr river flow, all SSCs at NPP plain are completely safe. River flow would start spreading over the fields on the left side, downstream of the dike and would start going backwards, in the "upstream" direction, behind the NPP.

At flows of about 1000-yr river flow value, the water would start to overflow the Potočnica creek dike. All NPP SSCs would be completely safe.

At the 10000-yr 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 side 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 side.

At flow corresponding to the current estimate of PMF in the plant licensing basis, 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, 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 CS 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 HP Safety Injection 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.

Evaluation of external flooding margin

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.

River flow approaching 3290 m3/s (100-yr Flood).

All SSCs at NPP plain are completely safe. Toward the upper end, the river flow would start spreading over the fields on the left side, downstream of the dike. Normal operation 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.

River flow between 3290 m3 and 4040 m3/s (1000-yr Flood).

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.

River flow between 4040 m3/s and 4790 m3/s (10000-yr Flood).

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 NPP 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.

River flow between 4790 m3/s and 6500 m3/s (PMF in Plant Licensing Basis).

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 theADP-1.03.30 which would, in this case, mean the operation of AFW 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.)

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.

River flow > 7081 m3/s.

It is not possible, currently, 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 El. 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-yr 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. Refer, also, to the conclusions which follow.

Conclusions regarding the margins for external flooding

As noted earlier, 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. 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-yr and 10000-yr 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 yr or larger. From that point cliff edge effect can be defined at this flow which is about 1.7 higher than existing PMF flow.

Containment integrity

Regarding the margin for containment integrity, analogous evaluation was performed to the one done for the seismic margin in section 2.2.

As before, 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.

River flow approaching 4790 m3/s (10000-yr Flood).

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

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-1.03.30) 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.

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 uncover and) 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 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.

Spent fuel pit

Spent fuel pit elevation permits to conclude that the effects of flooding will only affect support systems as for other structures and systems as discussed above. Time to fuel uncovery would allow for the implementation of some of alternative ways to achieve heat sink, described in the EOP and SAMGs.

Recapitulation of discussed flooding flows and levels

With the exception of historical values for the original design, table 6 recapitulates the flooding flows and levels of interest appearing in the different discussions.

Table 6: Flooding flows and levels

DESCRIPTION	FLOW	LEVEL
	m3/s	m.a.A.s.
100 year flood	3290	
Plant alert	3765	
1000 year flood	4040	
Site emergency	4274	
10000 year flood (design flood)	4790	155,35*
Buildings openings		155.50
Cooling tower exhaust stop logs		155.70
Probable maximum flood	6500	155.89*
Cooling tower water canal		155.90
Probable maximum flood plus wave	6500	156.35
ANSI/ANS-2.8-1992 flood scenarios (revised PMF)	7081	156.41
Current dike level at plain		157.10
Estimated cliff edge	11000	

* at dike

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

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 plant 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 CS 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 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.

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). This will further increase plant safety.

4 Loss of electrical power and loss of ultimate heat sink

4.1 Nuclear power plant

4.1.1 Loss of off-site power

4.1.1.1 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 gross. 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 with neighboring switchyards with three high voltage transmission lines. One of the transmission lines is connected to the Elektro-Slovenija network junction Maribor and the other two transmission lines (Zagreb I and Zagreb II) are connected to the network of the neighboring country Croatia. The switchyard 400 kV busses are also extended to RTP Krško and connected to 110 kV switchyard via 400 kV/110 kV transformer. Two unit transformers are connected between generator load breaker and step-up transformers and provide normal onsite power supply to two Class 1E (MD1 and 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 powered through direct burial underground cable from 110 kV RTP Krško or directly from combined gas-steam power plant Brestanica. Brestanica is equipped with three gas powered units of 23 MVA (23 MW) capable of blackstart in the event of a breakdown of the 110 kV system and to provide electrical power to Krško NPP station auxiliary transformer in 20 minutes.

If the 400 kV switchyard bus protection system is activated, the substation breaker will open, the turbine function Load Drop Anticipator will be activated and plant will be powered from generator through closed generator load breaker.

If transformer and generator protection system is activated, the substation breaker will open, the turbine will trip and will cause the reactor trip. The fast transfer of the 6.3 kV busses to the station auxiliary transformer powered from 110 kV RTP Krško will be automatically performed. If this fast transfer is not successful, the undervoltage on each of the emergency 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 emergency busses in less than 10 seconds. Safety related equipment will automatically connect to the emergency 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 on each of the emergency 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 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 of three gas turbines in Brestanica power plant is issued in accordance with instructions. The power to the 110 kV switchyard RTP Krško from Brestanica power plant can be established in 20 minutes. There is a possibility to establish power directly from Brestanica PP to the Krško NPP by closing the Q92 switch which bypasses the 110 kV switchyard RTP Krško.

4.1.1.2 Autonomy of the on-site power sources

If off-site 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. Emergency diesel generators are cooled by air and do not need other systems for cooling. Each diesel has underground reservoir with minimum 85 m3 of fuel allowing 85 hours of operation on full power. The required power to one train of safety systems is approximately 1.5 MW which means at least 7 days of emergency diesel generator operation are allowed for this load.

4.1.1.3 Provisions taken to prolong the time of on-site 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 m3) and the fuel for auxiliary boilers can be transferred to the underground reservoirs.

4.1.1.4 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 emergency bus which can be connected to either one of the existing emergency buses.

4.1.2 Loss of off-site power and on-site back-up power sources

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

4.1.2.1.1 Battery capacity and duration

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.

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).

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. However with the multiplication of additional diesel generators (one fixed and six mobile), 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, Lighting Panel, DC panel for control of 400 kV substation breaker 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.

4.1.2.1.2 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 off-site and on-site power:

1. In seismically qualified condensate storage tanks there in enough water for removing the decay heat through both steam generators.
2. 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.
3. Safety batteries capacity ensure power to 118 V instrument power supply.
4. Opening doors ensure appropriate temperature in turbine driven auxiliary feedwater pump room and in main control room cabinets.
5. Containment isolation can be done with locally closing isolation valves.
6. With local actions to isolate letdown line the inventory loss is minimized.

Safe plant condition can be prolonged by the use of alternative equipment which can be connected to the various systems through installed connection points in less than one hour.

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

4.1.2.1.3 (External) actions foreseen to prevent fuel degradation

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:

1. 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 m3 (200,000 gallons). 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 m3 (227,190 gallons).

   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.

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

   If condensate storage tanks are not available, then water flow can be established directly to the suction of turbine driven auxiliary feedwater pump by the same alternative methods.

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

   The guidance is provided for establishing alternative power from 150 kW portable diesel generator 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 to the pressurizer relief tank via the letdown line relief valve exists.

3. 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.

   The power can be established also to the lighting distribution panel to establish normal lighting in the main control room.

   Another important load powered from LD11 is positive displacement pump 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 batteries charger B can be established also by energizing motor control center MCCD211 with one of the three 150 kVA portable diesel generators. Batteries charger B can provide the power to the DC distribution panel train B and charge the 125 V batteries B and through inverters 2 and 4 the instrumentation distribution panels 2 and 4. DC distribution panel train A can be powered through swing charger A – B.

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.

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

   This action will minimize RCS leakage, prevent deterioration of RCP seals and permit makeup of water from SI accumulators to the reactor coolant system. 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 at 15 kp/cm2 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. Fire track reservoir can be filled with water from demineralized water tanks, fire protection tank, from main condensers, circulating water tunnel or from river Sava.

   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 8 kp/cm2. 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 river Sava.

   It is noted that the primary coolant loss is rather low and rapidly decreases when 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.

5. To enhance the equipment cooling, certain doors are open:
   - doors of main control room cabinets,
   - door of turbine- driven auxiliary feedwater pump
   - doors between intermediate building and turbine building.

6. In case of inadequate spent fuel cooling the operators are instructed to initiate spent fuel pit makeup using alternative equipment as described in section 4.2.2 of this report.

7. 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 5.2.1 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.

On-site 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 on-site sources for diesel or petrol fuel are not enough for long-term operation of this alternative equipment, the external delivery will be necessary.

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

Two petrol driven 125 V aggregates are 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 are available for possibility to remove decay heat in early stage after reactor shutdown and depressurizing steam generators.

4.1.2.2 Loss of off-site power and loss of the ordinary back-up source, and loss of any other diverse back-up source

Krško NPP possesses diverse back-up sources of power supply as described in section 4.1.2.1.

In case those backup electrical sources could not be connected at the buses level for any reason due to unavailability or no access, strategies for feeding at lower levels or directly to the load connection boxes are available and can be accomplished. Similarly for water sources and water injection strategies more than one alternative is available to cope with unavailabilities and difficulties on access for different scenarios.

4.1.2.2.1 Battery capacity and duration

N/A

4.1.2.2.2 Autonomy of the site before fuel damage

N/A

4.1.2.2.3 (External) actions foreseen to prevent fuel degradation

N/A

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

N/A

4.1.3 Loss of the ultimate heat sink

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

The Essential Service Water System 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.

The component cooling system 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 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 off-site power and any single active or passive failure.

Essential service water is supplied to a minimum of one component cooling heat exchanger; it is available to one electric auxiliary feed water pump or to the turbine driven auxiliary feedwater pump; the system operates during any plant normal or accident condition and during a safe shutdown earthquake with the loss of off-site electric power and any single failure, thus satisfying the safety function and single failure criterion required for this system.

The Essential Service Water System is classified as a Safety Class 3 and Seismic Category I system. The safety-related, pressure retaining components and parts conform to the requirements of the 1971 edition of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, and all addenda up to and including the winter 1972 addenda. In addition, the safety related components are designed to conform to the design limits and operability requirements of NRC Regulatory Guide 1.48, Design Limits and Loading Combinations for Seismic Category I Fluid System Components.

A low dam across the Sava River is used to maintain the water level at a nominal elevation of 150 m. However, the Essential Service Water System 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 0C (80 0F) and a minimum of 0.6 0C (33 0F).

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 Essential Service Water System 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 essential service water 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 essential service water 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 travelling water screen. The two channels are connected through two 0.91 m (36 inch) openings located on the inlet side of the trash racks and discharge to a common forebay serving the three pumps. The inlets, channels and travelling 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 travelling screen wash systems is supplied from the essential service water loops. Water is supplied through self-cleaning strainers to horizontal, centrifugal pumps which then deliver it to the travelling 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 essential service water 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 off-site 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 essential service water side of the Component Cooling Heat Exchangers. Four normally closed safety-related isolation valves provide isolation between the essential service water 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 essential service water loop, including screen wash system, associated travelling 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 off-site power sources. In the event of a loss of off-site power, each bus is supplied by separate, on site emergency diesel generators.

The essential service water 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. 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 essential service water piping.

A portion of the essential service water 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 m3/h (3000 gpm) 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 (36 °F).

The operation of the essential service water 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 off-site 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 essential service water. Correspondingly, if a failure occurs in the component cooling operating loop and operation is switched to the alternative loop, the essential service water backup loop must be started to provide cooling water to the new operating loop.

Normal operation of the travelling 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 travelling 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 Essential Service Water System.

The essential service water 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 two meters (6.5 feet) below the plant grade elevation of 155.20 m. This locates the pipe below the frost line and provides sufficient earth coverage for protection from surface loads.

4.1.3.2 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 River Sava.

If the flow from essential service water 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
- Verify both pressurizer spray valves are closed
- To minimize loss of reactor coolant through the reactor coolant pumps seals it is important to establish seal injection flow with positive displacement pump and to isolate letdown flow.
- Check the reactor coolant temperature and try to control the temperature.
- Establish boration flow to the reactor coolant
- Establish feed flow in to both steam generators with turbine driven auxiliary feedwater pump or one main feedwater pump
- Establish control of level and pressure in the pressurizer
- Establish instrument air with condensate polishing compressor (cooled with pretreatment water)
- Verify that reactor coolant is cooled by natural circulation
- Maintain stable plant condition with control of feed flow to both steam generators and with steam release into the atmosphere or condenser if available.
- Increase the makeup of demineralized water to have enough capacity to fill condensate storage tank, or potable water can be used as alternative source of water.
- Based on information of cause of the malfunction of essential service water, actions need to be started to restore the essential service water.

4.1.3.2.1 Autonomy of the site before fuel damage

"No load" temperature is maintained with natural circulation of reactor coolant, feeding both steam generators with feedwater and steaming with steam generator PORV(s). Auxiliary feedwater turbine driven pump normally provides feed flow to both steam generators, alternative main feedwater pump can also be used. Steam generator PORV(s) 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 has to be set to minimum.

Inoperable essential service water system for longer period will require the plant cooling down 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 turbine driven auxiliary pump and also enough steam pressure to remove decay heat from the core with releasing steam from steam generators.

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: demineralized water storage tanks, fire protection tank, condenser, circulating water tunnel, river Sava, potable water from city of Krško.

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

Increased water level in PRZR can be compensated with decreasing injection flow to the RCP with adjustment of the seal injection flow to the minimum value of 1,6 m3/hr per RC pump. After a long period of time the level in the PRZR will be increased, temperature on primary side will be maintained between 130 and 150 °C and pressure will be around 20-25 kp/cm2. To prevent solid operation it will be necessary to start decreasing level in PRZR. Low pressure letdown flow cannot be used due to inoperable RHR system (loss of component cooling). Level in PRZR can be decreased with PRZR PORV into the PRT. Also, normal letdown flow can be used in case of bypass of CVCS demineralizers.

Source of water for PDP pump is RWST, it will be necessary to prepare BAT for filling with boric acid.

4.1.3.2.2 (External) actions foreseen to prevent fuel degradation

a. 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 (OPC). Organization for emergency situation is established according to the (NZiR) and EIP procedures (Emergency Implementation Procedures). Plant well water or portable water from Krško city can be used for different purposes such as filling the condensate storage tanks, reactor water storage tank, filling the spent fuel pit, boric acid tank, reactor makeup water storage tank and for preparation of additional demineralized water.

b. 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.

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

a. Operation of centrifugal charging pump and high head safety injection pump is important to provide enough flow to have capability to maintain level and pressure under control. For operation of charging and safety injection pump it is important to establish some coolant for the oil coolers, so is important to have some kind of component cooling available.

   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 River Sava 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, high head safety injection pump and even the auxiliary feedwater 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 component cooling system and essential service water system.

b. Alternative cooling can also be established with installation of 8" tee on the existing 24" SW line to CC 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 m3/hr which can provide enough heat removal for one train of Emergency Core Cooling System (ECCS) and also to remove decay heat from spent fuel pit.

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

d. New water line from Krško hydro plant which could be installed in the near future could provide alternative way of cooling the component cooling heat exchanger. Gravity force will be used as passive cooling system.

4.1.3.3 Loss of "primary" heat sink and "alternative heat sink"

N/A

Krško NPP does not have an »alternative heat sink«; all actions are described in section 4.1.3.2.

4.1.3.3.1 Autonomy of the site before fuel damage

N/A

4.1.3.3.2 (External) actions foreseen to prevent fuel degradation

N/A

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

N/A

4.1.4 Loss of the primary heat sink, combined with station black-out

4.1.4.1 Autonomy of the site before fuel damage

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

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

Decay heat removal is achieved with turbine driven auxiliary feedwater pump and steam relief into the atmosphere through steam generator PORV(s). For the first four (4) hours (or more as per 4.1.2.1.1), there are batteries and compressed nitrogen in bottles to operate valves (steam generator PORV(s) and control valves for turbine driven auxiliary feedwater pump). During that period alternative source of power and compressed air can be established or we can manually control the speed of turbine driven pump and manually release the steam from steam generator 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 steam generators.

For the alternative power supply we can use one of two portable diesel generators, which are located on the highest 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.

The power can be established also to the lighting distribution panel to establish normal lighting in the main control room.

Another important load powered from LD11 is positive displacement pump 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 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 batteries charger B can be established also by energizing motor control center MCCD211 with one of the three 150 kVA portable diesel generators. Batteries charger B can provide the power to the DC distribution panel train B and charge the 125 V batteries B and through inverters 2 and 4 the instrumentation distribution panels 2 and 4. DC distribution panel train A can be powered through swing charger A – B.

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.

During the loss of the primary heat sink component cooling heat exchanger cannot be cooled and consequently spent fuel pit cannot be cooled; as a result we cannot use the equipment which uses component cooling water as cooling media for operation.

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 turbine driven auxiliary feedwater pump. 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.

All alternative portable and mobile equipment is located on-site at least 100 yards away from the reactor. 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.

In case turbine driven auxiliary feedwater pump is not available portable fire protection pumps can be used to supply water into both steam generators. These pumps have enough capacity to remove the decay heat from the core and to maintain the level in both steam generators to provide natural circulation on primary side.

4.1.4.2 (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 on-site will be used. All necessary actions can be performed with shift crew and additional personnel from Technical Support Center (TSC) and Operating Support Center (OPC). Organization for emergency situation is established according to the Protection and Rescue Plan (NZiR) and EIP procedures (Emergency Implementation 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 condensate storage tank, demineralized water storage tanks, potable water, well water an also the River Sava. External support can also be provided by enough water capacity from Krško potable water source or any other available water source.

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

- 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.
- Provision to cool the component cooling heat exchanger passively (gravity feed from Krško hydro power plant).
- Alternative residual heat removal could be established with skid mounted pump and heat exchanger and connection points to RH system.

4.2 Spent fuel pit

4.2.1 Loss of off-site power

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

The spent fuel pit 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 spent fuel pit cooling system water is drawn from the spent fuel pit by one spent fuel pit pump and is pumped through the tube side of one heat exchanger, and returned to the spent fuel pit. 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 manually.

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 spent fuel pit cooling system is capable of maintaining the spent fuel pit water temperature at or below 72.4 °C (162.3 °F) when the spent fuel pit 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 (164.3 °F).

Spent fuel pit cooling pumps are powered from safety related 400 V busses LD11 and LD12. In the event of a loss of off-site power the safety-related emergency buses can be powered either from 110 kV Krško Distribution and Transformer Station (TS) through station auxiliary transformer or from emergency diesel generators as described in section 4.1.1.

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

4.2.1.2 Autonomy of the on-site power sources

The autonomy of the emergency diesel generators is described in section 4.1.1.2.

4.2.1.3 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 4.1.1.3.

4.2.1.4 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 4.1.1.4.

4.2.2 Loss of off-site power and of on-site back-up power sources

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

4.2.2.1.1 Battery capacity and duration

The battery capacity and duration is described in section 4.1.2.1.1.

Spent fuel pit temperature and level instrumentation is normally powered from the process information system. In case of loss of normal power, this instrumentation can be transferred to the dedicated battery.

4.2.2.1.2 Autonomy of the site before severe accident

If loss of all AC power occurs, spent fuel pit cooling pumps will be lost and the cooling flow to the spent fuel pit heat exchangers will be lost. The temperature of water in spent fuel pit will start increasing with rate 4.6 °C/hr at heat generation 8.36 MW. If the initial temperature is conservatively 73.5 °C then the time to boiling is 5 hours and 37 minutes. Heat removal from the spent fuel is now established by water boiling in the spent fuel pit. The mass of evaporated water is 3.7 kg/s which means water level decrease of 10.2 cm/hr. For maintaining the constant water level in the spent fuel pit it is required to deliver water flow at least 13.5 m3/hr. If water is not delivered into the spent fuel pit, then the USAR limit 7.1 m of water is reached in 46.6 hours and the beginning of uncovering the spent fuel elements is at 74 hours after event initiation.

4.2.2.1.3 (External) actions foreseen to prevent fuel degradation

Procedures instruct the operators to monitor the spent fuel level and temperature and to initiate the makeup to the spent fuel pit.

If the power to the bus LD11 is established as described in section 4.1.2.1.3, then the normal makeup to the spent fuel pit can be established from refueling water storage tank through purification lines or from reactor makeup storage tank.

Alternative means for establishing spent fuel pit makeup:

- Pumping water from water pretreatment tanks with portable fire pump to the system for purification of spent fuel pit water surface.
- Providing water from fire protection hydrant network to the system for purification of spent fuel pit 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 carbonate mud pool with portable fire pump to the system for purification of spent fuel pit water surface.
- Pumping water from circulating water intake pool with submersible fire pump and fire track to the system for purification of spent fuel pit water surface.
- Pumping water directly to spent fuel pit from fire protection system.

If water in the spent fuel pit is decreasing even if makeup to the spent fuel pit 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, carbonate mud pool, circulating water intake and circulating water outlet pool.

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

Installation of fixed piping above the spent fuel pit with connections for portable fire pumps.

4.2.2.2 Loss of off-site 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 supply other than these described in section 4.1.2.1.

In case those backup electrical sources could not be connected at the buses level for any reason due to unavailability or no access, strategies for feeding at lower levels or directly to the load connection boxes are available and can be accomplished. Similarly for water sources and water injection strategies more than one alternative is available to cope with un availabilities and difficulties on access for different scenarios.

4.2.2.2.1 Battery capacity and duration

N/A.

4.2.2.2.2 Autonomy of the site before severe accident

N/A.

4.2.2.2.3 (External) actions foreseen to prevent fuel degradation

N/A.

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

N/A.

4.2.3 Loss of the primary heat sink

4.2.3.1 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 Pit Cooling and Cleanup System (SFPCCS) is designed to remove the decay heat generated by spent fuel assemblies stored in the spent fuel pit. 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 (124.3 °F) when the heat exchanger is supplied with component cooling 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 (125.2 °F). The flow through the spent fuel pit 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 spent fuel pit cooling system is capable of maintaining the spent fuel pit water temperature at or below 72.4 °C (162.3 °F) when the spent fuel pit 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 (164.3 °F).

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

The spent fuel pit 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 spent fuel pit cannot be drained more than four feet below the normal water level (normal water level is approximately 7.3 m (24 feet) 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 spent fuel pit 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.

"Alternative means for establishing spent fuel pit makeup:

- Pumping water from water pretreatment tanks with portable fire pump to the system for purification of spent fuel pit water surface.
- Providing water from fire protection hydrant network to the system for purification of spent fuel pit 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 carbonate mud pool with portable fire pump to the system for purification of spent fuel pit water surface.
- Pumping water from circulating water intake pool with submersible fire pump and fire track to the system for purification of spent fuel pit water surface.
- Pumping directly to spent fuel pit with hoses from fire protection system.

If water in the spent fuel pit is decreasing even if makeup to the spent fuel pit 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."

SFP level measurement covers all 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 Spent Fuel Pit.

4.2.3.2 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 spent fuel pit is performed by the Krško NPP operating personnel who are normally on shift. All actions are performed according to the operating procedures.

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

An alternative system with skid mounted pump and heat exchanger to cool the spent fuel pit.

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

4.2.4.1 Design provisional autonomy of the site before severe accident

Heat removal from the spent fuel pit can be achieved basically in two ways; i.e. through spent fuel pit heat exchanger or through evaporation of water from pent fuel pit 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 spent fuel pit and there is no concern about criticality.

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

Source of water can be provided from different tanks located at the plant, portable water, well water or water from the River Sava 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 spent fuel pit. This could be done to the skimmer connection through valve 13030 or directly with the use of fire protection hoses into the spent fuel pit.

Until 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. If water in spent fuel pit is decreasing, even if make-up to the spent fuel pit is established, than 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 the portable fire protection nozzles ensures adequate cooling of spent fuel elements.

In case of losing the level of the spent fuel pit there would be no criticality concern.

Cooling of spent fuel pit is provided by evaporation in case of loss of ultimate heat sink. To verify that the cooling is adequate, temperature and level need to be monitored.

Instrumentation is provided to measure the water temperature in the spent fuel pit, 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 spent fuel pit and give an alarm in the control room when the water level in the spent fuel pit reaches either the high or low level set points (15 cm above or 16 cm below the normal water level (115.00).

All actions will be performed by a shift crew on-site based on guidance of emergency operating procedures.

4.2.4.2 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 on-site would be used.

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

a. Alternative cooling can also be established with installation of 8" tee on the existing 24" SW line to CC 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 m3/hr which would provide enough heat removal of the decay heat from spent fuel pit.

b. New water line from Krško hydro plant which will be installed in the near future will provide an alternative way of cooling the component cooling heat exchanger. Gravity force would be used as a passive cooling system.

c. Installation of 3 or 4" piping with isolation valve in vicinity of FHB door. Piping should be routed into the spent fuel pit, which will provide flow directly into the spent fuel pit. With this arrangement operators are protected against radiation even in the case of low water level in the spent fuel pit.

d. An alternative system with skid mounted pump and heat exchanger to cool the spent fuel pit.

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

4.2.5.1 Design provisional autonomy of the site before severe accident

Design provisional Autonomy of the site before severe accident is described in section 4.2.3.1.

Loss of the primary heat sink, combined with station black-out 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 spent fuel pit can be achieved basically in two ways; i.e. through spent fuel pit heat exchanger or through evaporation of water from spent fuel pit 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 spent fuel pit and there is no concern about criticality.

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

Source of water can be provided from different tanks located at the plant, potable water, well water or water from the River Sava or any other water. Deminerlized water or clean water without impurities would be preferred.

With the portable or mobile pumps with their own engines (independed from power source) water can be transported into the spent fuel pit. This could be done to the skimmer connection through valve 13030 or directly with the use of fire protection hoses into the spent fuel pit.

Until 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. If water in spent fuel pit is decreasing, even if make-up to the spent fuel pit is established, than 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 the portable fire protection nozzles ensures adequate cooling of spent fuel elements.

In case of losing the level of the spent fuel pit there would be no criticality concern.

For monitoring both the level and the temperature, temperature and level measurement is possible independently of the electrical power source.

Instrumentation is provided to measure water temperature in the spent fuel pit, and to give local indication as well as annunciation at the main control board when normal temperatures are exceeded. Alternative instrumentation is provided on-site (different power supply from battery or generator, local indication, wide range), which enables supervision of SFP temperature in the case of loss of AC power.

Instrumentation is provided to measure water level in the spent fuel and give an alarm in the control room when the water level in the spent fuel pit reaches either the high or low level set points (15 cm above or 16 cm below the normal water level (115.00).

There is also alternative equipment (different power supply from battery or generator, local indication, wide range) available on-site, which enables supervision of SFP water level in the case of a loss of spent fuel pool water without regard to its origin.

4.2.5.2 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 on-site would be used. All necessary actions can be performed with shift crew and additional personnel from Technical Support Center (TSC) and Operating Support Center (OPC). Organization for emergency situation would be established according to the Protection and Rescue Plan (NZiR) and EIP procedures (Emergency Implementation Procedures).

For long-term operation, external support would be 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 condensate storage tank, storage tanks with demineralized water, potable water, well water and also the River Sava. 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 on-site based on guidance in operating procedures

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

- Mobile water station with enough capacity to meet all needs for decay heat removal.
- Connection point for alternative cooling of component cooling heat exchanger.
- Provision to cool the component cooling heat exchanger passively (gravity feed from Krško hydro power plant).
- New water line from Krško hydro plant which will be installed in the near future will provide an alternative way of cooling the component cooling heat exchanger. Gravity force would be used as a passive cooling system.
- Installation of 3 or 4" piping with isolation valve in vicinity of FHB door. Piping should be routed into the spent fuel pit, which will provide flow directly into the spent fuel pit. With this arrangement operators are protected against radiation even in the case of low water level in the spent fuel pit.
- Installation of spray lines around the spent fuel pit with pipe lines to allow the possibility of spraying the spent fuel with enough water capacity to remove decay heat and to provide enough water to prevent fuel uncover
- An alternative system with skid mounted pump and heat exchanger to cool the spent fuel pit.

5 Severe accident management

5.1 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 also 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 local RERPs of municipalities Krško and Brežice, and RERPs of Posavje region and the Republic of Slovenia. Emergency preparedness is regularly coordinated on all levels.

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 the 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.

Krško NPP's EPZs are shown in Figure 9.

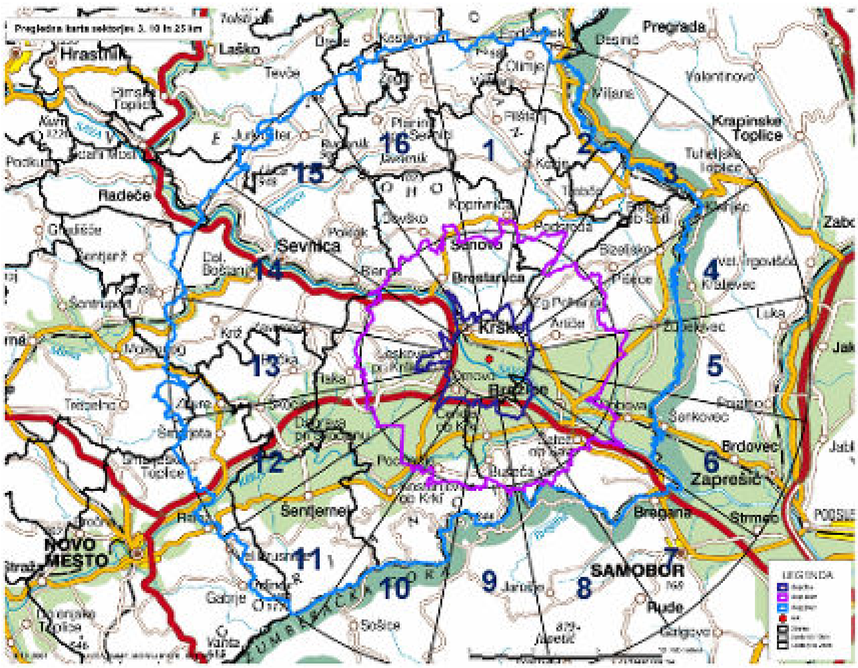

Figure 9: Krško NPP Emergency Planning Zones

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 on-site specific emergency action levels (EAL) derived from an example of initiation events for each level of emergency and methodology given in NUREG-0654 "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants", App. 1.

Emergency classification considers internationally accepted 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) and 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 which 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 on-site and off-site emergency response organization (ERO) to the extent required and initiation of appropriate accident mitigation and emergency response measures.

5.1.1 Organization planned

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

For maintaining the emergency preparedness, clear 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 Off-site 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 emergency preparedness planning. The particular accident management activities and emergency preparedness elements are conducted in different departments and through different plant programs – e.g. in Radiological protection through the Radiological environmental monitoring program, in Operations through the Fire protection program, in Training through training program, in Security through security programs etc.. These different activities and entire Krško NPP's emergency preparedness planning is in compliance with the plant’s Emergency preparedness program prepared by the Engineering services division.

Krško NPP’s accident management and emergency planning is in agreement with national legislation, European directives and US standards and regulation as well as other 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 plant's emergency preparedness and planning is determined in Krško NPP's RERP. Krško NPP's RERP is a license document, reviewed at least annually and approved by the Krško NPP management board. The main objectives of the RERP are:

- identification and evaluation of various types of accidents and emergencies which could potentially occur at the plant;
- identification of on-site accident management and emergency response measures;
- identification of on-site 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 off-site support;
- delineation of obligations of Krško NPP regarding off–site emergency response;
- delineation of coordination of response activities with off-site authorities;
- identification of Krško NPP's emergency facilities, equipment and communications;
- identification of on-site recovery measures and organization;
- provision of basic instructions for maintaining on-site emergency preparedness.

5.1.1.1 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 on-site emergency response and co-ordination with off-site emergency response authorities;
- effective realization of particular emergency response measures.

Krško’s ERO is based on normal internal operating organization and consists of organizational structures activated depending on the emergency level and located in emergency facilities. Organizational structure of the ERO (see Figure 10) is as follows:

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

b. on-site part of ERO (Technical support centre (TSC) and Operational support centre (OSC)), which is activated in alert or higher level of an emergency;

c. off–site 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 which is also the location of Civil Protection Headquarters of the Republic of Slovenia.

d. Additional support to the Krško NPP ERO is provided on contractual basis by Off-site support organizations regardless of the emergency level.

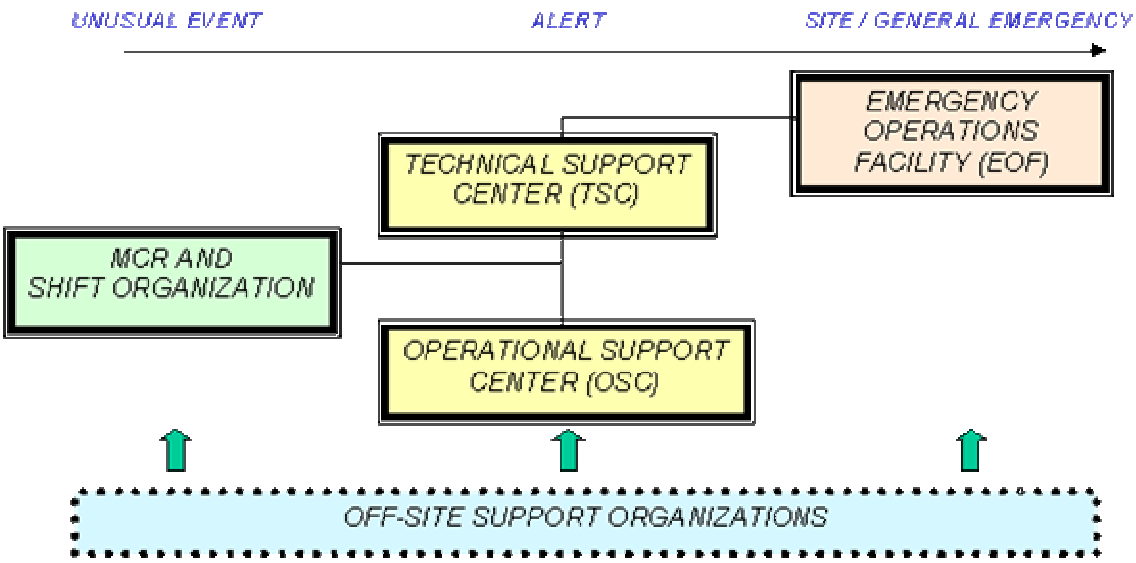

Figure 10: Krško NPP RERP organization

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.

The Krško NPP 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 clearly defined in the Krško NPP RERP and Emergency Implementing Procedures (EIP). In order to minimize confusion and assist in the control of the emergency response, the ERO is designed so that only one person, or his alternative, is responsible for the implementation of specific emergency actions. In addition, the functional areas of responsibility will 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 appropriate 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. This means practically the whole Technical division, Engineering division and Security Department and partially individuals from other plant organizational units.

Krško NPP 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 ERO activation drills. The response time of first essential intervention staff is 30 minutes after alerting, which is in compliance with recommendations of NUREG-0654, Table B-1. The TSC is able to carry out its emergency functions in about 1 hour after alerting. The Emergency Director must declare the TSC operable after conditions for effective TSC operation are fulfilled. The EOF is able to carry out its emergency functions in about 2 hours after activation. The EOF Director has to declare the EOF operable after conditions for effective EOF operation are fulfilled.

Krško NPP's ERO can be activated as follows:

1. Alerting individuals on mobile telephones using the software application (RR) 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 on their time 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 when personnel is gathering 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 RR is checked every day. The response of ERO is checked monthly and is not announced in advance.

5.1.1.1.1 Staffing and shift management

In the case of unplanned shutdown or reactor trip, predetermined technical and engineering personnel are activated for further coordination of corrective actions and repairs. This group is not considered as part of the ERO.

In the case of an unusual event, the following positions are also activated: 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 organization is shown in Figure 11.

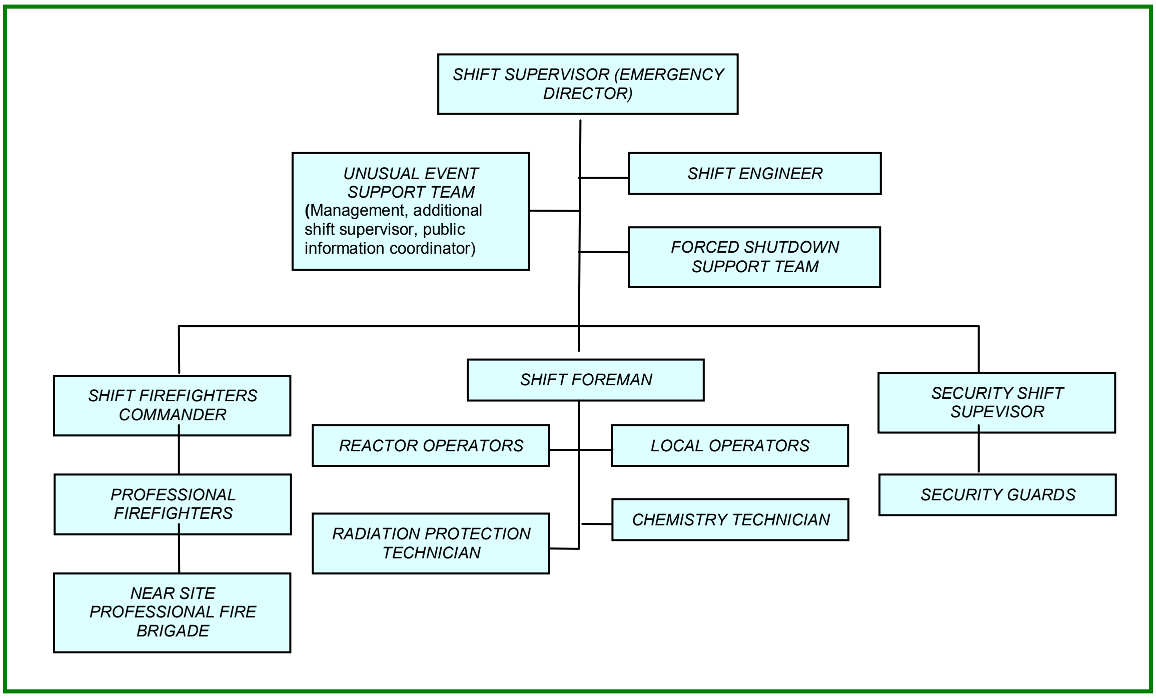

Figure 11: Krško NPP ERO organization

Until the TSC is operable, management and coordination of the emergency response is organized within the MCR shift. 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 an adequate individual 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 clearly 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 located inside the plant. 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;
- on-site 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 off-site support organizations and authorities;
- evaluation and making decisions on severe accident management strategies;
- coordination of logistics;
- assuming 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 assumes the position of EOF director until this position is established. The clear 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;
- on-site technical and engineering support coordinator regarding the plant and core status evaluation, TSC operation, technical and engineering evaluations and support, emergency classification, off-site notifications, evaluations of severe accident management strategies;
- radiation protection coordinator regarding control of radiation exposure and dosimetry, radiological survey, evaluation of off-site radiological consequences and coordination of off-site 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 12.

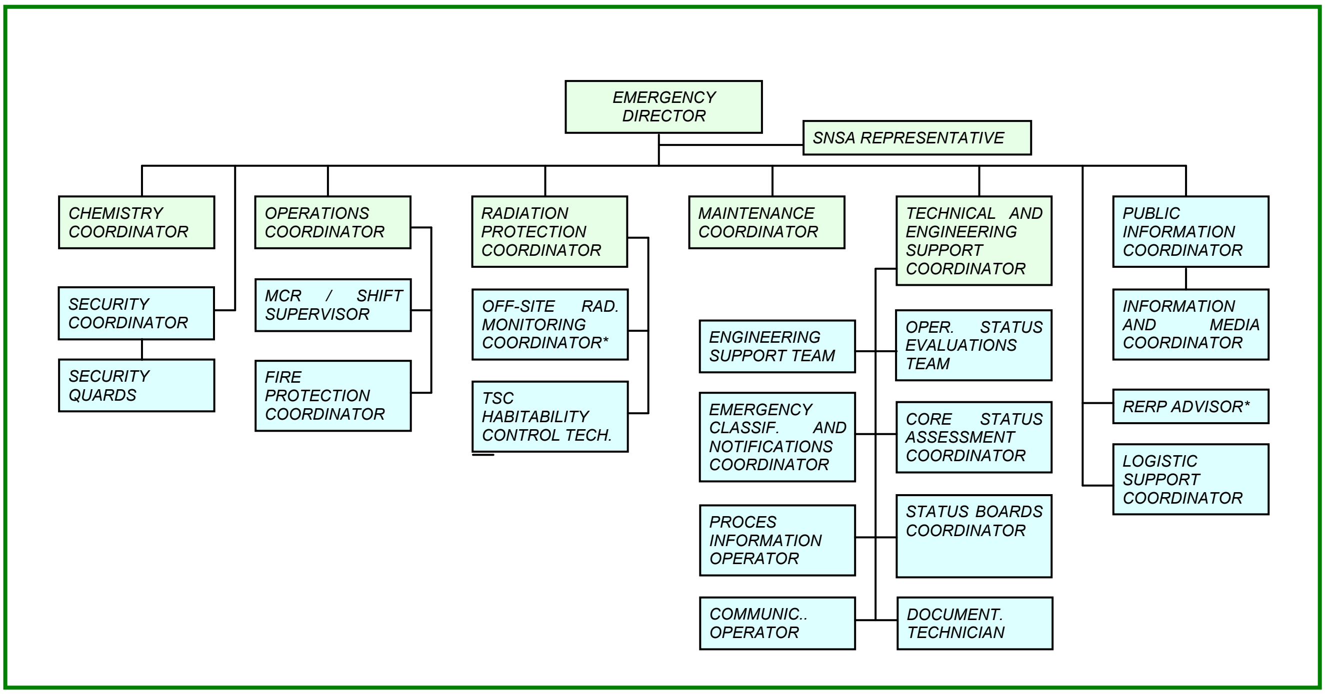

Figure 12: Krško NPP RERP organization

Operational Support Center (OSC) organization

The OSC is organized, equipped and structured to place the intervention teams and carry out the on-site 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 13.

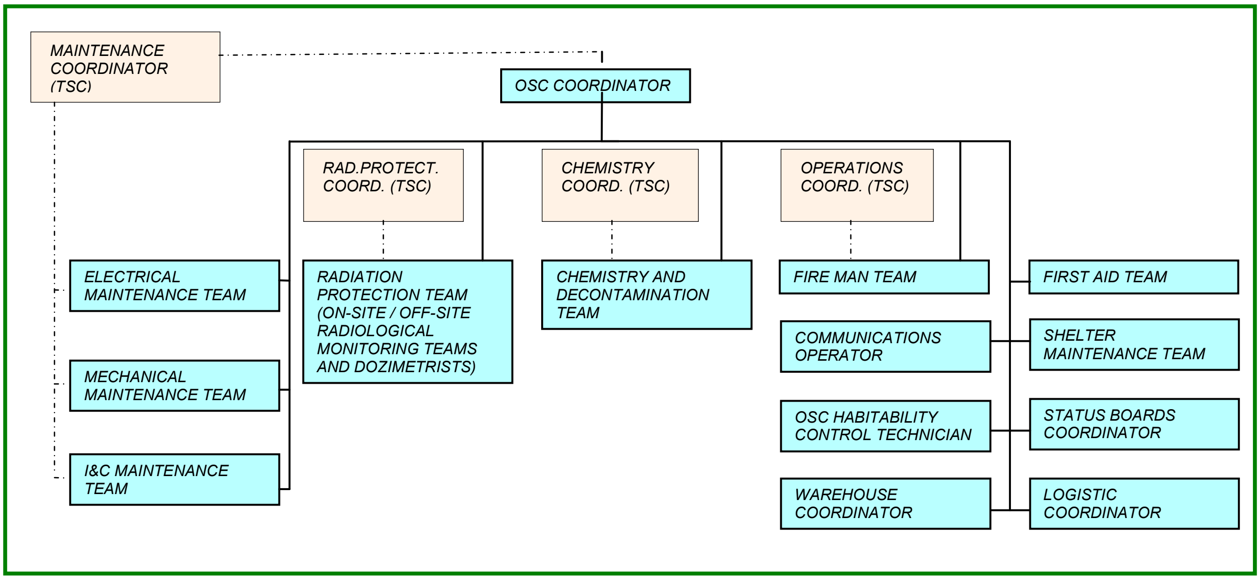

Figure 13: Krško NPP OSC organization

If TSC and/or OSC primary locations are not available, TSC/OSC is evacuated to the other on-site locations or to the near-site alternative location (AL). The AL 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 AL 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 Offsite 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 on-site;
- coordination with the Civil Protection Headquarters of the Republic of Slovenia, the SNSA and off-site support organizations;
- emergency classification and off-site notifications;
- evaluation of off-site 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 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 resources to manage on-site emergency. Non-delegated responsibilities of the EOF Director are emergency classification, off-site protective actions and off-site authorities’ notifications. He is responsible for recovery measures after the emergency close-out. The clear 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 on-site emergency response;
- off-site dose assessment coordinator regarding evaluation of off-site radiological consequences, off-site urgent protective actions and coordination of these activities with off-site organizations;
- EOF engineering and technical support coordinator regarding engineering and technical support, EOF operation, emergency classification, notifications and coordination with off-site authorities (SNSA);
- public information coordinator regarding public information activities, rumour control, media coordination and coordination of these activities with off-site organizations;
- logistic coordinator regarding logistic support;
- other support personnel in EOF regarding implementation of individual 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 14.

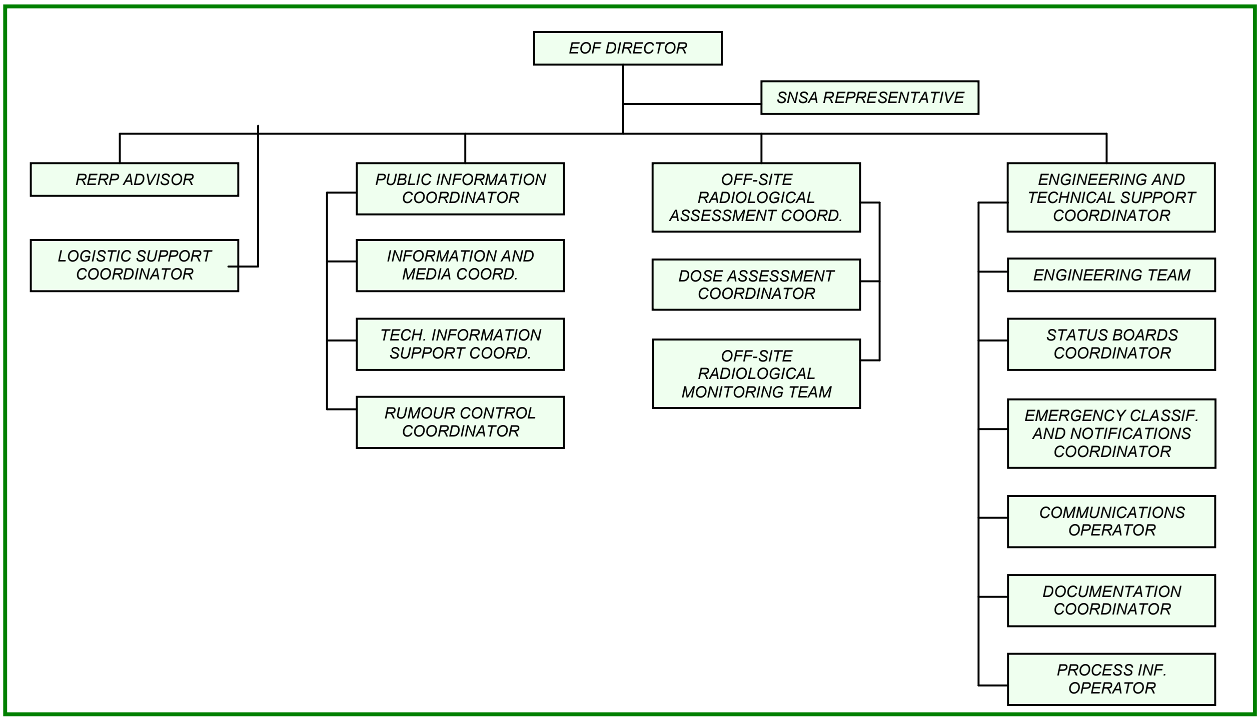

Figure 14: Krško NPP EOF organization

5.1.1.1.2 Measures taken to enable optimum intervention by personnel

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

The ERO intervention teams (including operators and security guards) are man-powered for shift turnovers during the interventions and 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 are educated, trained and well-prepared for their emergency tasks. Coordination, preparation, direction and execution of duties in the area of different emergency response are regularly a part of the drills and exercises. 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 clearly 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.

Krško NPP considers regulation requirements in the RERP and emergency response procedures and respects them during emergency management.

5.1.1.1.3 Use of off-site technical support for accident management

Off-site support and assistance to Krško NPP is provided by the local and other off-site 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 off-site 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 off-site support for the following emergency response duties:

- 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 concluded.
- Providing intervention medical assistance to victims of non-radiological or radiological injuries and all medical centre capabilities 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 concluded.

- 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 operation facility (EOF). Written notice with the national authority provided.
- Providing transmitter location for the NPP's off-site radiological monitoring radio communication channel. Contract with local radio club is renewed annually.
- Providing location for TSC and OSC near site alternative emergency location (AEL). Long-term contract with the local institution is renewed annually.
- Providing on-site 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 contract is maintained in the Krško NPP Fire protection plan. The professional fire brigade assures off-site support any time in 5 to 10 minutes after alerting. However, in case the fire brigade is hindered due to eventual catastrophic events, Krško NPP has on-site 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.
- Ensuring priority separate (isle) off-site power supply to NPP from the nearest Gas Power plant in case of loss of other off-site power. Agreement and operating instructions with relevant companies in the Republic of Slovenia.

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.

5.1.1.1.4 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. The structure of procedures is user-friendly. 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 and beyond design basis accidents 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 emergency implementing procedures (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. Emergency response training, drills and exercises are planned within annual emergency response training plan.

The following participants are involved in emergency response training: all Krško NPP's employees and contractors' personnel, intervention personnel assigned to the Krško NPP ERO and intervention personnel of off-site support organizations. Krško NPP employees and contractor personnel become familiar with the RERP and emergency protective measures within the frame of regular general employee training.

The intervention personnel assigned to the Krško NPP ERO including operator and security guards receive training for their respective assignments. Emergency response training for these personnel consists of initial, continuing (prequalification) and specialized (proficiency) emergency response training. The initial emergency response training is conducted for the individuals upon their assignment to the ERO. Both the RERP and emergency procedures are wholly involved in this training. Continuing emergency response training is conducted for ERO personnel annually and contains a general review of the RERP with the emphasis on individual themes. Specialized emergency response training is a supplement to the continuing emergency response training and includes specific themes regarding the functions and tasks individuals have in ERO. For example, fire-fighters, maintenance intervention teams, local operators have specialized training in manipulation with on-site mobile equipment; operators and plant status evaluation team have specialized training in EOPs and SAMGs backgrounds etc. The intervention staff of the off-site support organization also participates in emergency response training program.

Krško NPP regularly conducts 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.

The Krško NPP is carrying out the drills with the frequency as follows:

- off-site 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;
- off-site radiological monitoring – three times per year;
- assessment of off-site 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 – three times per year (all evacuees participate in evacuation drills);
- activation of the ERO – once per year (the response of the intervention personnel is checked once per month);
- use of on-site severe accident mobile equipment and preparation for severe accident management based on EOPs or SAMGs strategies (on yearly basis).

An emergency response exercise is carried out annually to evaluate overall emergency response readiness of Krško NPP and participating organizations and integration of emergency response segments. 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 mobile equipment for managing severe accidents training is provided to adequate personnel from Fire brigade, Maintenance and Operations department and to the Emergency Response Organization personnel. The fire fighter and shift personnel will perform 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 drills are opportunities FOR OTHER Emergency Response Organization personnel to obtain training on mobile equipment.

The full-scope real time Krško NPP’s simulator serves as a scenario simulation tool for technological transients and accidents (including severe accidents). The simulator is also used for the real MCR simulation.

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 (SCBA) in simulator control room and evacuation of simulator control room.

5.1.1.2 Possibility to use existing equipment

Krško NPP has most of the equipment to manage an accident and emergency on-site. 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 off-site 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 (HF) 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 HF 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, 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 off-site radiological monitoring tasks. In case of severe radiological conditions, essential radiological instrumentation can be transferred from on-site to an off-site clean location.

The types and quantities of on-site 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.

Most of the maintenance tools and instrumentation are placed in the central workshop in the administrative building. Some tools are also available in the hot workshop in the radiological control area and on other locations on-site. Spare parts are stored in two on-site warehouses and in the warehouse located 500 m from the plant.

Most of 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. Most of the equipment is adequately stored and maintained. The intervention staff knows the equipment location and is regularly trained on using it. The access to most of the equipment is also possible in accident conditions or in case of loss of power.

5.1.1.3 Provisions to use mobile devices (availability of such devices, time to bring them on-site 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 off-site support and up to 1 week with no needs for additional heavy mobile equipment from off-site. The mobile equipment essential for managing severe accidents (SAME) according to EOP and SAMG strategies are stored at different locations on-site. 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 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.

In Table 7 is a list of on site available mobile equipment (SAME).

Table 7: Krško NPP Severe Accident Management Mobile Equipment (SAME)

KRŠKO NPP SEVERE ACCIDENT MANAGEMENT MOBILE EQUIPMENT (SAME)		
1.	Portable generator AE900AGR-001	0.4 kV / 5 kW
2.	Portable generator AE900AGR-002	0.4 kV / 5 kW
3.	Portable generator AE900AGR-003	0.23 kV / 2.6 kW
4.	Portable generator AE900AGR-004	0.23 kV / 2.6 kW
5.	Portable oil free compressor AE900CPR-001	1620 m^3/h / 10.3 BAR
6.	Portable oil free compressor AE900CPR-002	1620 m^3/h / 10.3 BAR
7.	Mobile diesel generator AE900DSL-001	0.4 kV / 600 kVA
8.	Mobile diesel generator AE900DSL-002	0.4 kV / 1000 kVA
9.	Mobile Diesel generator AE900DSL-004	0.4 kV / 150 kVA
10.	Mobile Diesel generator AE900DSL-005	0.4 kV / 150 kVA
11.	Mobile Diesel generator AE900DSL-006	0.4 kV / 150 kVA
12.	Portable fire protection pump AE900PMP-001	50 kW / 60 m^3/h / 15 BAR
13.	Portable fire protection pump AE900PMP-002	50 kW / 60 m^3/h / 15 BAR
14.	Submersible pump AE900PMP-003	2.8 kW / 60 m^3/h / 1 BAR
15.	Submersible pump AE900PMP-004	2.8 kW / 60 m^3/h / 1 BAR
16.	Submersible pump AE900PMP-005	2.8 kW / 60 m^3/h / 1 BAR
17.	Submersible pump AE900PMP-006	2.8 kW / 60 m^3/h / 1 BAR
18.	Trailer with HS60* HIGH PRESS AE900PMP-008	240 m^3/h / 3 BAR
19.	Portable transformer AE900XFR001	230/118 V / 3 kVA
20.	Portable transformer AE900XFR001	230/118 V / 3 kVA

5.1.1.4 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 m3 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 on-site 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 off-site water supplies. A 5-ton boric acid supply on-site is maintained permanently. It is estimated that the stock of 3000 KI tablets is enough to protect on-site intervention personnel for a period of 2 weeks. Krško NPP maintains stores of other supplies (food, drinking water etc.) to function without off-site 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.

5.1.1.5 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 filters (HEPA) 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:

1. 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.
2. The containment spray system has a dual function:

   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.

   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.
3. 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.

4. 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;

<u>Design of major containment components</u>

1. 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.
2. 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.
3. The reinforced concrete shield building which provides the required biological shielding between the containment and outside spaces.
4. Double barrier penetrations at all pipe entrances which ensure prevention of air leakage around pipes while maintaining independent behaviour between the shield building and the containment vessel.
5. Annulus negative pressure control system is designed to limit the maximum pressure in the annulus immediately after loss of coolant accident (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.
6. Annulus filter system is designed to minimize off-site 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:

   Two 100-percent-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-percent-capacity exhaust fans are located in the auxiliary building.
7. 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.
8. 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 s design basis event. The recombiner is manually controlled from a panel located outside the containment.
9. 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 pit 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:

1. Plant vent header monitoring and isokinetic sampling for normal and accident conditions;
2. Secondary side condenser air ejector monitoring and sampling;
3. Steam generator leakage monitoring;
4. Containment high-range radiation monitoring;
5. Failed fuel monitor at reactor coolant let-down pipe;
6. Steam generator relief valve monitors;
7. Auxiliary building ventilation exhaust monitor and sampling;
8. Fuel handling building ventilation exhaust monitor and sampling;
9. 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 main control room. 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 Programme 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 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 x 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.

5.1.1.6 Communication and information systems (internal and external)

See section 5.1.2.1 below.

5.1.2 Possible disruption with regard to the measures envisaged to manage accidents and associated management

5.1.2.1 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 4000 m south of the site and local road Krško - Drnovo - Brežice 3000 m 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 2.000 m 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 (Main Control Room (MCR), Technical Support Center (TSC), Operational Support Center (OSC) …)
- 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 Off-site Facility (EOF) and other external organizations involved in the Emergency Preparedness Plan:

- Radio System (portable radio stations),
- Direct phone to the Slovenian Nuclear Safety Administration – 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.

All of these systems are powered from the different UPS's, 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 - SBO) on-site. 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 On-site buildings and Off-site 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 Off-site Emergency Facility 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 Systems in the OEF. The usage of Process Information System (PIS) in the OEF 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 (Technical Support and Operations Support Centers).

The PIS system on-site is supplied from non-safety 220 V DC batteries via three different Uninterruptible Power Supply 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 Main Control Room Staff.

The Plant operating staff excluding security (there are 15 technical individuals available all the time on-site) is able to implement the Plant Operating Emergency Procedures and other required actions from the Site Emergency Preparedness plan by itself without additional support for at least 24 hours. All equipment listed to be used during a serious event is present on-site with the following supplies:

1.	Emergency Diesel Generators	Fuel oil supply for 7 days of operation
2.	Additional Portable Emergency Equipment	Fuel supply for 72 hours of operation

However, in accordance with the Emergency preparedness plan the time to activate and to achieve the operability of the Plant Emergency Support Centers (Technical Support and Operational Support Center) is 1 hour and 2 hours for the Off-site Emergency Facility. The number of the Operational Support Center 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 on-site (3 persons).

5.1.2.2 Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities on site

5.1.2.2.1 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.

1. 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.
2. To adequately meet the requirements of 10CFR Part 50, Appendix A, General Design Criterion 19, to permit access and occupancy of the Main Control Room under the postulated accident conditions.

3. To provide a Control Room Emergency Charcoal Cleanup System for operation under abnormal conditions.
4. 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.
5. 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:

1. Main Control Room Air Conditioning System
2. Main Control Room Charcoal Clean-Up System
3. Chilled Water Generating and Distributing System

During post-accident conditions, pneumatically operated dampers automatically isolate control building rooms from the outside atmosphere. The main control room air conditioning and electrical room cooling systems assume full recirculation. The main control room cleanup system is manually started, if required, during control room isolation to keep the area habitable.

The control room isolation is initiated by either a safety injection signal or by a high radiation signal as detected by the radiation monitor in the main control room, 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 main control room, relay and switchgear rooms, 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 (UHS) 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 ºC. 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 on-site, together with diesel-powered charging compressor.

If the Main Control Room 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.

5.1.2.2.2 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 action plan includes requirements to mitigate the consequences of accidents in which reactor core is severely damaged. It is related specifically to the item "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems which may be used in post accident operations". 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.

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.

The ventilation system of the TSC and OSC are both equipped with different type of filters including HEPA and Charcoal filters.

5.1.2.3 Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods)

The emergency preparedness (EIP) is a constituent element of the entire Krško NPP's nuclear safety concept and an integral part of the plant's internal organizational structure and working process (Ref. Krško NPP Emergency Preparedness). For the implementation of this extensive plan there are several different facilities (in addition to the Main Control Room) equipped to host the emergency preparedness staff. In accordance with the NPP EIP, the following locations (centers) are established:

- Technical Support Centre (TSC),
- Operational Support Centre (OSC)
- Off-site Emergency Facility in Ljubljana (EOF).

The TSC is recognized as a coordinating center of operation, maintenance and other activities during an emergency. The Main Control Room 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 Main Control Room could be used. In the case of the plant inaccessibility, the Alternative Emergency Location (AEL) 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.

5.1.2.4 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 on-site, independent power source which consists of two diesel generator units and the DC battery systems, designed with sufficient capacity to furnish on-site 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 steam turbine driven auxiliary feedwater pump. 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 on-site 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 5.1.2.5).

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 24 hours and above (see 4.1.2.1.1) 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.

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. 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 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.

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 ESF 400 V 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 on-site for diesel generators’ 3-day operation. It takes approximately 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 (MCCD)s by simply plugging the ready cable into the socket, which is mounted on the wall near the MCCD. 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 on-site. With provisions on the systems with universal firefighting connections installed, they can pump water from virtually any tank on-site, 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 on-site, 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 on-site. 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-volt 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 control room, 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 control room, 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** AELS 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.

5.1.2.5 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 or Engineered Safety Features Actuation System. 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 5.1.2.4 there are alternative means and methods available to enable the operation of at least limited instrumentation channels.

5.1.2.6 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 southeast 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 northwest 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 the smaller Cerklje Airport located 8 km to the south from the site. The airport is used only for the military services and few private light aircraft traffic.

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 on-site. 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 on-site. 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 Fighting Brigade 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 Emergency operating procedures 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 to provide adequate prevention from all known 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.

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.

5.2 Nuclear power plant

5.2.1 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

5.2.1.1 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 loss of coolant accident (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). 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.

Engineered safety features (ESF) 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 systems, subsystems, and components are provided to satisfy the above cited functions and are designated as engineered safety features:

1. Containment Systems
2. Emergency Core Cooling System
3. Control Room Habitability System
4. 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:

1. 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.
2. 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.
3. 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.
4. 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.

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 loss of coolant accident 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 loss of coolant accident conforms to the following criteria:

- Peak cladding temperature will not exceed 1200 ºC.
- Cladding oxidation will not exceed 17% of the total cladding thickness.
- Hydrogen generation (due to zirconium-water reaction) will not exceed 1 percent 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, as presented on Figure 17, 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.

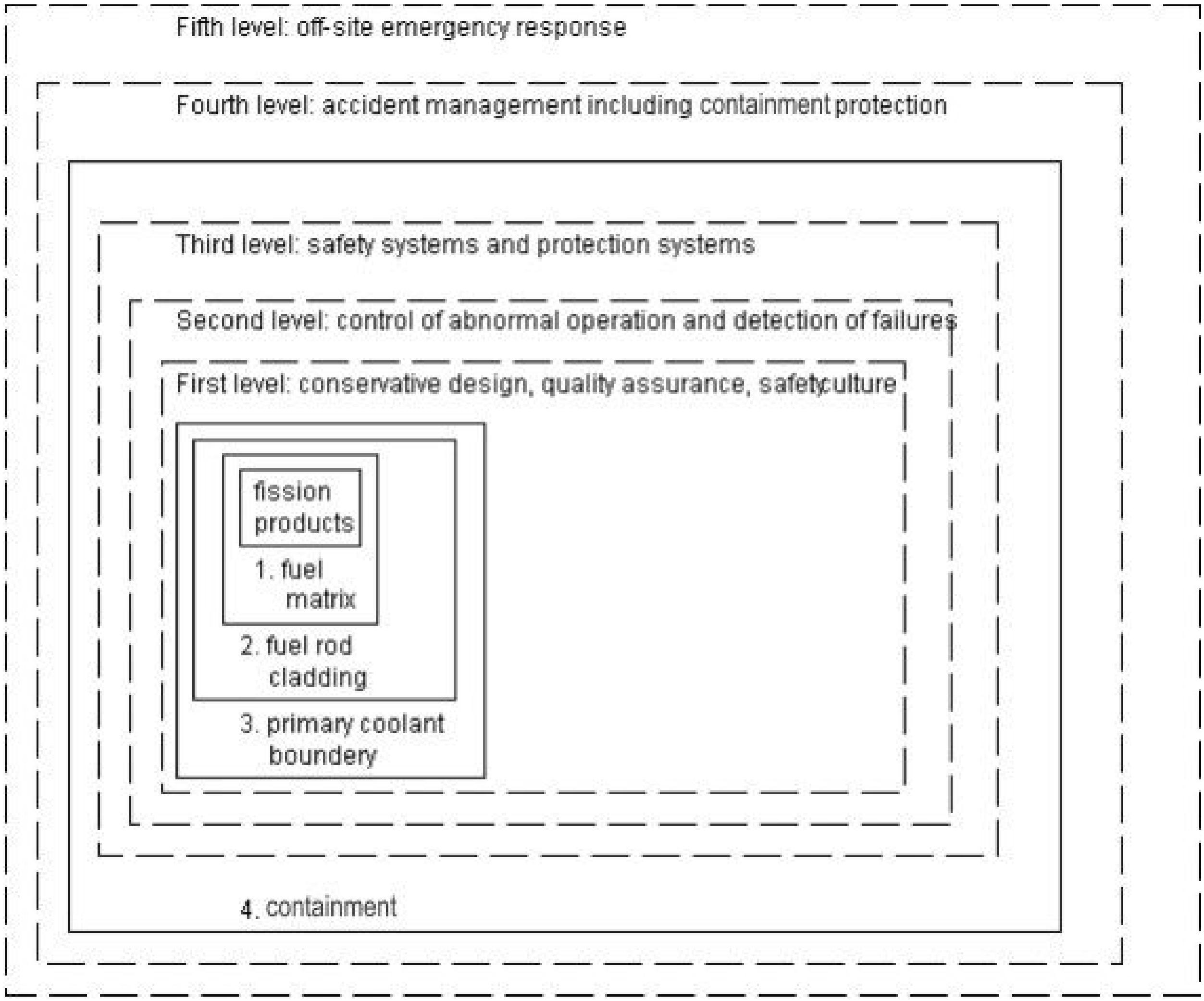

Figure 15: Defense in depth concept presented graphically

5.2.1.1.1 Preventive measures

Given the existence of the automatic engineered safeguards systems (ESF), the emergency operating procedures, 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. Main Control Room (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 18 is showing operators response to abnormal situation by using different level of procedures.

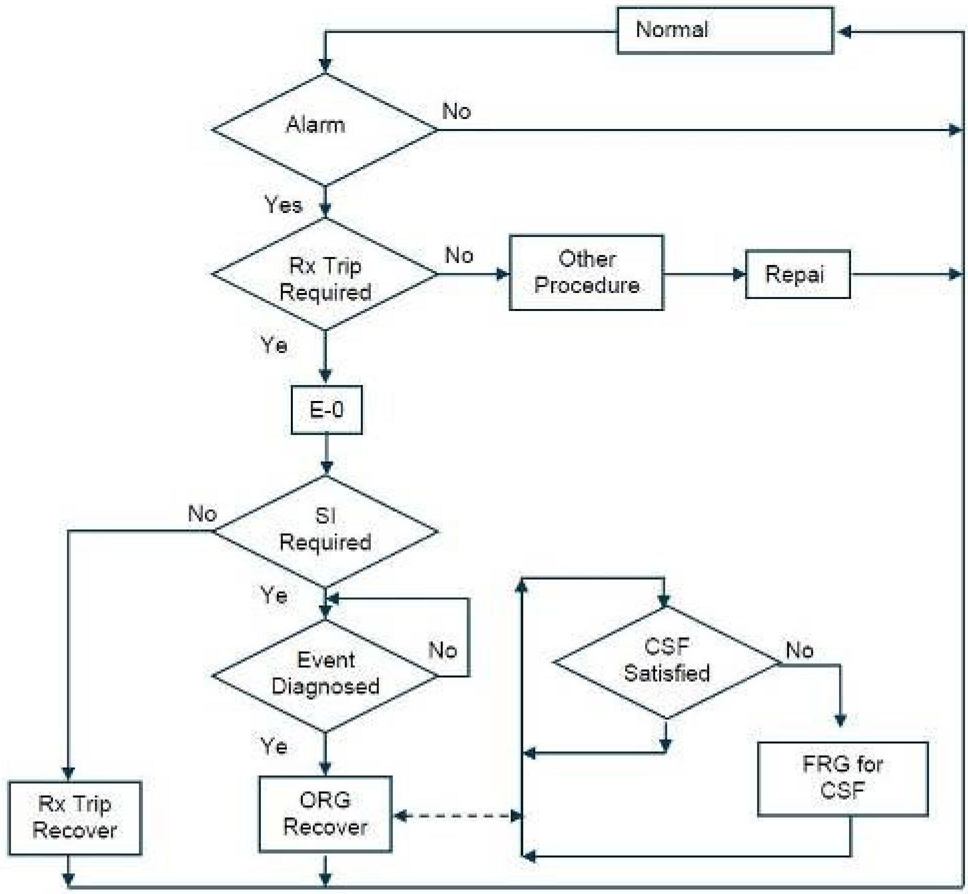

Figure 16: Emergency response guideline usage

Alarm Response Procedures (ARP) guide operators to take proper action in their response to main control room 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 main control room 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 18).

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, reactor coolant system 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, reactor coolant system 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. Reactor Coolant System Integrity (preventing failure of RCS),
5. Containment Integrity (preventing failure of containment vessel),
6. Reactor Coolant Inventory (preventing flooding and loss of pressure control).

Relation between CSFs and barriers is shown on Figure 19.

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.

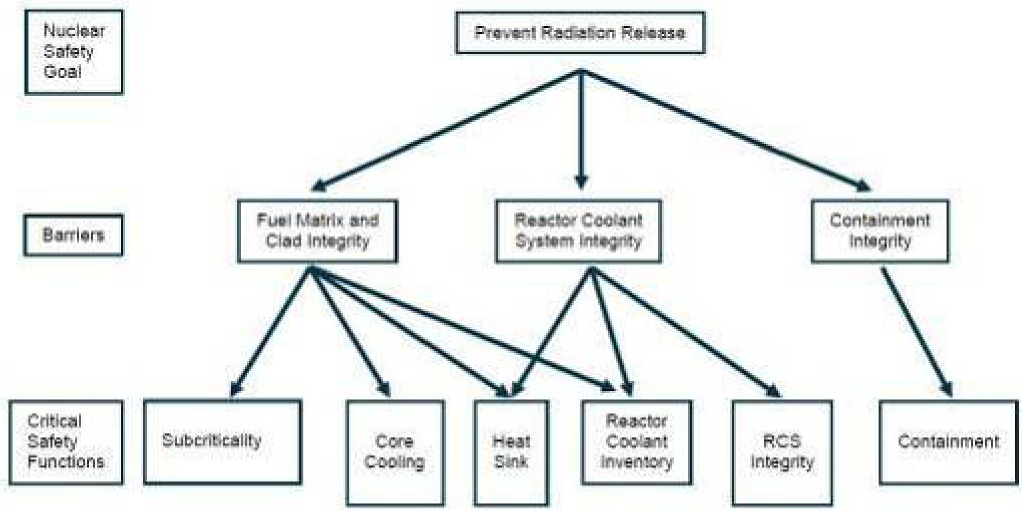

Figure 17: Relation between FRG's goal, barriers and critical safety functions

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.

5.2.1.1.2 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 20.

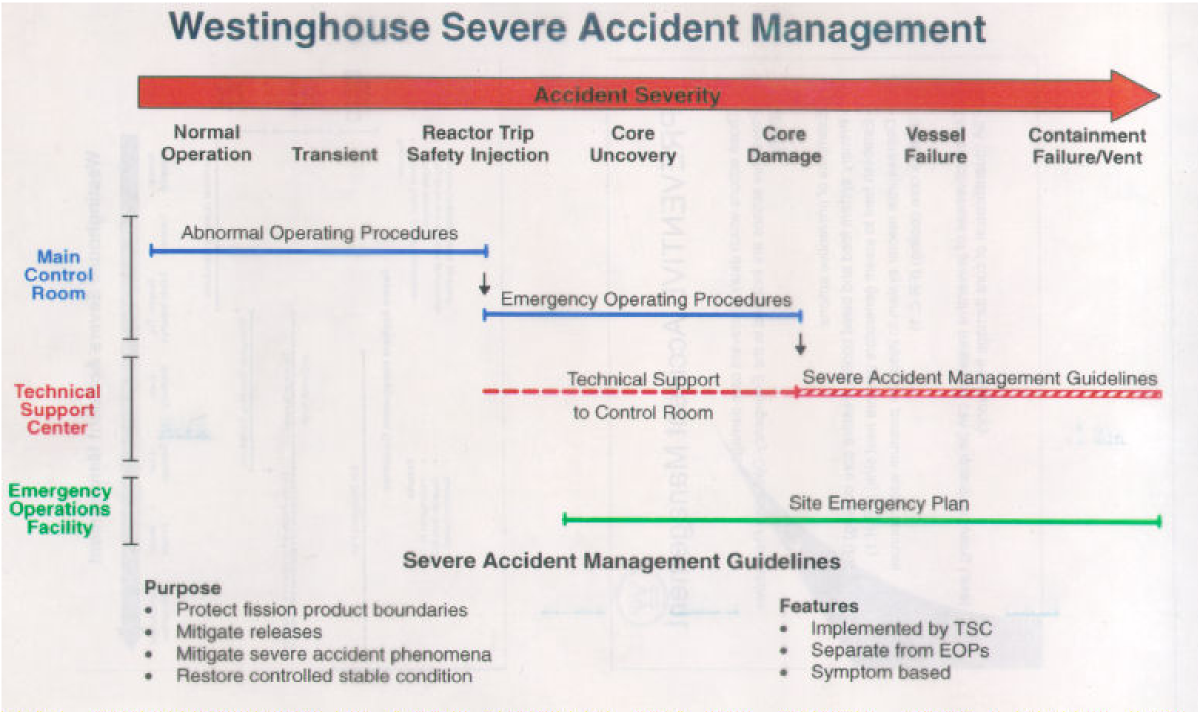

Figure 18: Graphic structure of Krško NPP procedures and guidelines in case accident scenario is leading to core damage

Krško NPP performed analyses where long-term Station Blackout (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 Turbine Driven Auxiliary Feedwater Pump, 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” Emergency Operating Procedure. Reactor coolant pumps (RCP’s) 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 off-site 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 steam generators (SG) using power operated relief valves, and this would lead to the primary side cooldown and depressurization. RCS pressure decrease results in passive accumulator injection to reactor coolant system which lasts until RCS pressure drops below pressure when accumulators should be isolated. Turbine driven auxiliary feedwater (AF) pump 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 regulator valves are operable (if nitrogen or alternative compressed air supply is available) and that condensate storage tanks can be refilled by variety of sources (water treatment tanks, pretreated water tanks, fire protection tanks, condenser hotwell, city water, circulating water tunnel, river Sava water), the secondary side heat sink would be available during the whole transient. Even if turbine driven AF pump 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 steam generators and variety of water sources available. Fire truck 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 auxiliary feedwater and emergency diesel generators. This is in accordance with the high importance of secondary heat sink function and with the high contribution of Loss of off-site power and Station blackout initiators to core damage probability.

Therefore Krško NPP has a variety of alternative methods for injecting water into both steam generators in the case of SBO together with mobile diesel generators for providing electrical power to particular equipment (battery charger, PDP, particular motor-operated valves). This mobile equipment is placed on-site and has fuel available for three days of operations on-site 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.

5.2.1.1.3 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 (ATWS event).

Loss of all AC power (Station Blackout - SBO)

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 quickly, 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 reactor coolant pump (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 on-site. 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 on-site 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.

Response to inadequate core cooling

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 steam generators power operated relief valves. 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 reactor coolant pumps (RCP). 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 power-operated relief valves 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 power operated relief valve are examples of different ATWS events. The required operators' actions following identification of an ATWS event are the same but the reactor coolant system 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 emergency operating procedure. With reference to particular tasks, operator actions may be as follows:

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 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.

Injecting into the reactor coolant system (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.

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.

Depressurize the reactor coolant system

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 power operated relief valves (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 PORV's, 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.

Late AC power recovery

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 on-site, 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.

Establishing of containment sump recirculation

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.

5.2.1.2 After entering a severe accident situation following damage to the fuel or even the pressure vessel

5.2.1.2.1 Identification of the risks, cliff edge effects and kinetics of severe accidents

Krško NPP performed sensitivity analysis for different long-term Station Blackout (SBO) accident sequences, with focus on the containment pressure response following core damage. Accordingly, several scenarios were evaluated, with respect to the reactor coolant pump (RCP) seal leakage flow, availability of the turbine driven auxiliary feedwater pump 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". Reactor coolant pump 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 off-site power sources are not available, operators immediately start establishing an alternative power supply from one of two available mobile diesel generators (located on-site with 3 days fuel supply) for running charging positive displacement pump (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 (PRT) increasing the coolant loss until RCS pressure decreases below valve set-point. Operators would rapidly depressurize secondary side using steam generators (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 reactor coolant system until pressure drops below the limit when accumulators should be isolated. Turbine driven auxiliary feedwater (AF) pump 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 regulator valves are operable (by using nitrogen or alternative compressed air supply available on-site) and that condensate storage tanks 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 turbine driven auxiliary feedwater pump is not operable, Krško NPP has available portable firewater pumps on-site, with variety of injection flow-paths to steam generators and variety of water sources. Fire truck 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, reactor coolant system and the containment integrity are not endangered.

Therefore, no cliff edge effect can be expected if a secondary heat sink is available.

If operators open containment isolation valves on sump recirculation lines by using on-site 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 truck), which can also effectively use reactor coolant pump 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.

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 truck) 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.
- 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 of SBO and loss of ultimate heat sink. 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 pit 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 emergency core cooling system failure) to several hours (e.g. a secondary side initiated transient, small breaks in the primary system or reactor coolant pump 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 UO2 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 centre 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 reactor coolant system 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 reactor coolant system 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 reactor coolant system 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 tube sand 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 he 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 (DCH) in the Krško NPP has been performed. Wet cavity design of Krško NPP would significantly mitigate the consequences of DCH.

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, has 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 molten core concrete interaction in the reactor cavity was investigated. The accidents are initiated respectively by a large break loss of coolant accident (LOCA), a small break LOCA, and a small break LOCA on the reactor coolant pump 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 pressurizer relief tank). 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. Reactor coolant pump 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 pit inventory

If spent fuel pit (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 zirconium-water reaction would be possible resulting in generation of more heat and producing hydrogen.

Hydrogen production is not expected until top of the fuel elements is uncovered. Time to reach the limit depends on total heat power of the fuel elements stored in the pool. If bounding case is considered with core unloaded from the reactor immediately after 18-months fuel cycle completion, it would take approximately 3.4 days to uncover fuel elements.

5.2.1.2.2 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 21.

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.

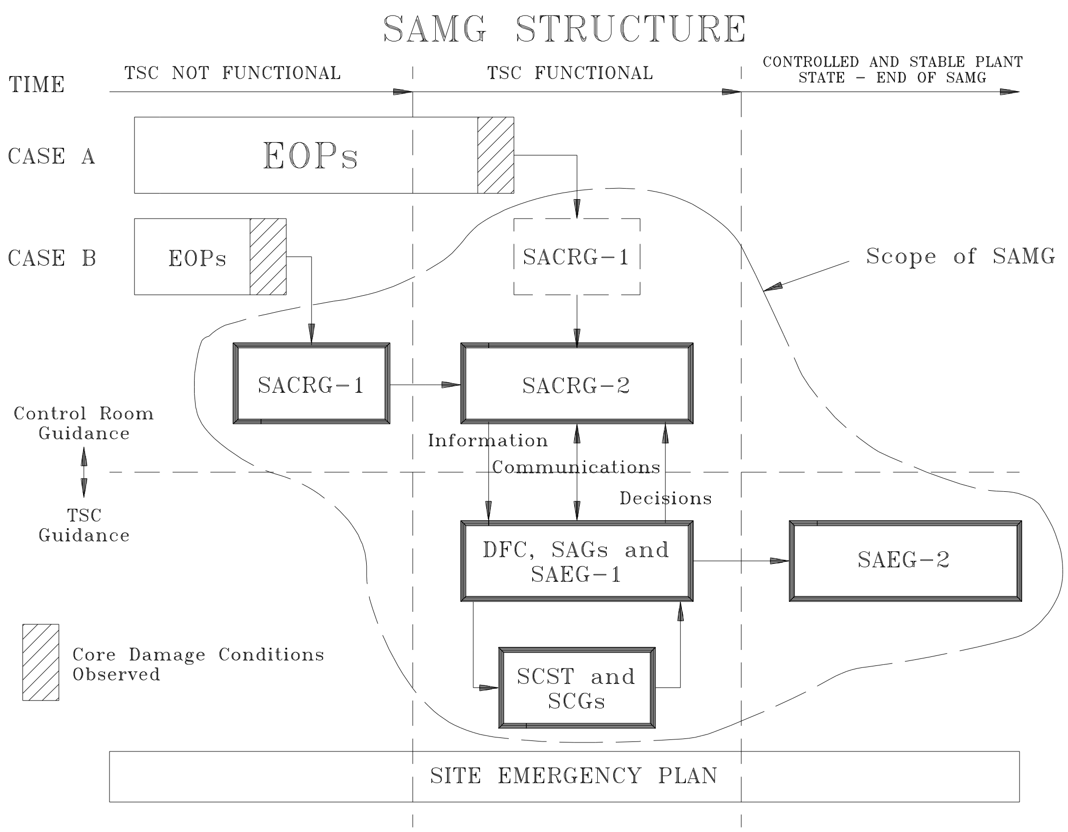

Figure 19: Graphic presentation of transition from Emergency Operating Procedures to Severe Accident Management Guidelines (before and after Technical Support Center is functional) and Severe Accident Management Guidelines structure

In the event of core damage accident there are three major types of response actions:

1. control and termination of fission product releases
2. prevention of severe challenges to the containment fission product boundaries
3. recover core cooling

Secondary actions are to:

1. Minimize fission product releases while achieving primary goals;
2. Maximize equipment and monitoring capabilities while achieving the primary goals.

The Westinghouse Owners group (WOG) SAMG 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 (plant evaluation team) 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 5.2.1.1.1.). 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 diagnostic flowchart (DFC), the TSC implements the corresponding Severe Accident Guidelines (SAG), taking into account orders of priority.

A list of all Krško NPP SAGs is in Table 8. Prioritization of SAMG strategies is shown on Figure 22.

Table 8: List of Krško NPP Technical Support Center Severe Accident Guidelines

TSC Guidelines and Associated Diagnostic Flowchart (DFC) Parameter		
GUIDELINE	**DESCRIPTION**	**Diagnostic Flowchart (DFC) Parameter**
SAG-1	Inject Into Steam Generators	Water Level in all SGs
SAG-2	Depressurize Reactor Coolant System	RCS Pressure
SAG-3	Inject Into Reactor Coolant System	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

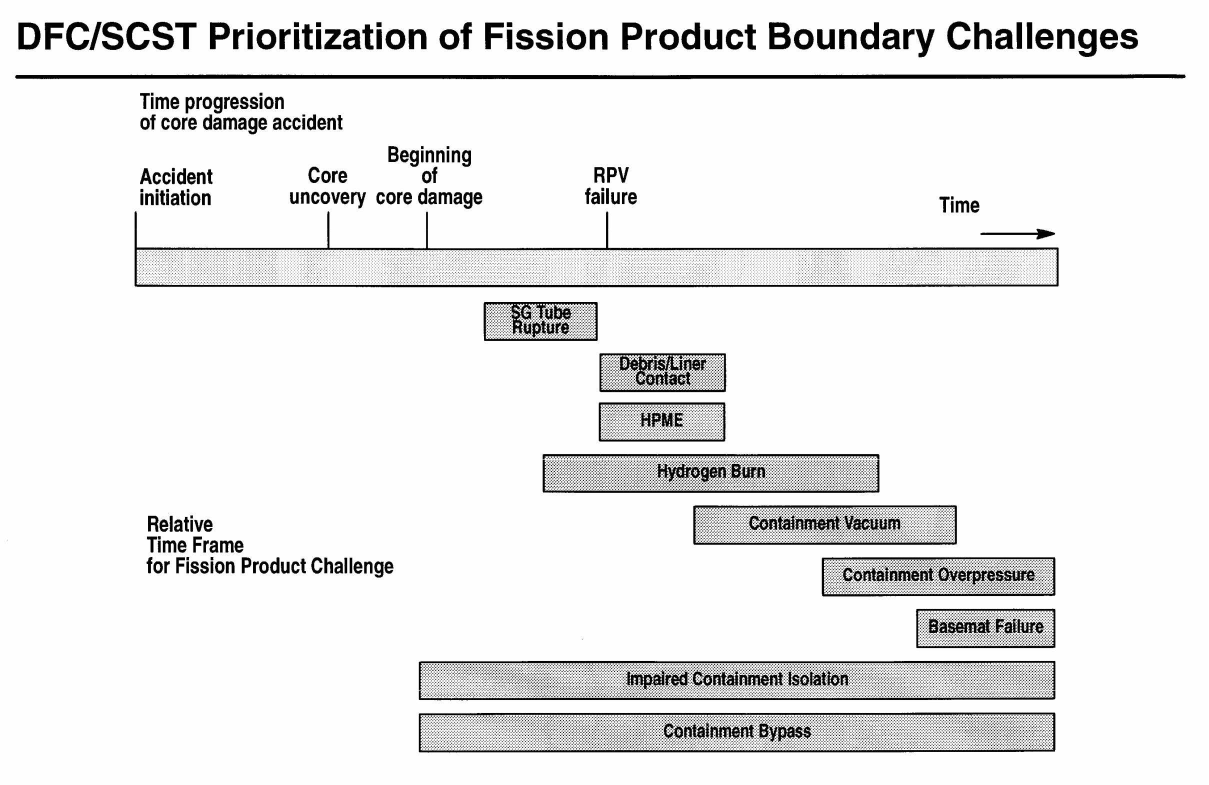

Figure 20: Diagnostic prioritization of SAMG strategies based on prioritization of fission product boundary challenges

Inject into steam generators (SAG-1)

The purposes of injecting into the steam generators are to protect the steam generator tubes from creep rupture (preventing breaching containment fission product barrier), to scrub fission products that enter the steam generators (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), on-site mobile firewater pumps and submersible firewater pumps for the river Sava 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 river Sava 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 steam generator 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, steam generator 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 steam generators once core exit temperatures exceed 650ºC, 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 molten core concrete interaction, 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 on-site with fuel supply available for three days of operation. Fire protection truck is available on-site 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 steam generators (through a tube leak or rupture), to the auxiliary building (through emergency core cooling system (ECCS) break located outside containment), or to the containment annulus.

Another concern is a loss of spent fuel pit (SFP) cooling. During prolonged loss of SFP cooling a loss of SFP inventory and fuel uncovery is expected.

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 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 containment fan coolers. If AC power is not available, portable fire protection pumps can be used for containment spray, 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 Steam Generator 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 spent fuel pit 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 on-site, as well as from the river Sava. In addition to that, water can be sprayed over the pool, using the same pumps and firefighting nozzles, also located on-site. 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 spent fuel pit 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 spent fuel pit 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 containment spray and fan coolers, 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 on-site: fire truck tank, water treatment tanks, fire protection tanks, condensate storage tanks, pretreated water tanks, condenser hotwell, city water, circulating water tunnel, and the river Sava water. Additional flow-paths are also provided: fire protections hoses connections to containment spray discharge lines and to reactor coolant pump 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 recombiner; but it does not represent a real limit for recombiners overheating. In the setpoint calculation the upper limit of operability of the hydrogen recombiner 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 3. In this case, ignition sources, including electric recombiners, would be isolated.

Since Krško NPP median containment failure pressure is at 7.9 bar, with the peak pressure line at maximum 6.8 bar, 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 on-site), 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 containment spray header (also available by portable fire protection pumps), through discharge of ECCS pumps to the RCS, through containment spray and ECCS recirculation lines to the containment sump, and through fire protection lines for spraying reactor coolant pumps. On-site 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 on-site: fire track tank, water treatment tanks, fire protection tank, condensate storage tanks, pretreated water tanks, condenser hotwell, city water, circulating water tunnel, and the river Sava 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 9.

Table 9: List of Krško NPP Technical Support Center Severe Challenge Guidelines

TSC Guidelines and Associated Severe Challenge Status Tree Parameters (SCG's)		
GUIDELINE	**DESCRIPTION**	**Severe Challenge Status Tree 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 Severe Challenge Status Tree. 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 Diagnostic Flowchart 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 containment recirculation fan coolers, injection of water by containment spray system, 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 containment spray, 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 river Sava 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 off-site 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.

5.2.2 Accident management measures and installation design features for protecting containment integrity after occurrence of fuel damage

5.2.2.1 Management of hydrogen risks (inside and outside the containment)

5.2.2.1.1 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 U.S. 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 Severe Accident Management Guidelines (SAMG), 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 Severe Challenge Guideline (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 asses 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 Severe Accident Guideline (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.

Another potential source of hydrogen is located outside containment. If spent fuel pit (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 spent fuel pit 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 Refueling Water Storage Tank 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 on-site.

5.2.2.1.2 Risks of cliff edge effects and deadlines

In accordance with the Krško NPP plant-specific containment hydrogen distribution study, the results indicate that the duration of large H2 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 Severe Accident Management Guidelines (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, hydrogen production is not expected until top of the fuel elements is uncovered. 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 3.4 days to uncover fuel elements.

5.2.2.1.3 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, Abnormal Operating, Emergency Operating and Severe Accident Guidelines 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 on-site. 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.

5.2.2.2 Prevention of containment overpressure

5.2.2.2.1 Design, operation and organization provisions

There are three methods for decreasing the pressure in the containment: heat removal via component cooling water with containment recirculation fans, 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 severe challenge guideline (specifically above 5.1 kp/cm2). 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 containment recirculation fans. 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 on-site, 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 bellow 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 spent fuel pit 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 on-site 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 5.2.3.1, 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 containment recirculation fans 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 reactor coolant pumps, 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 on-site. These pumps are completely self-sufficient, thus being able to operate under station blackout conditions.

5.2.2.2.2 Risks of cliff edge effects and deadlines

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) cam 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 (as fire track). By doing this containment will not fail within 7 days.

5.2.2.2.3 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 on-site. 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.

5.2.2.3 Prevention of re-criticality

5.2.2.3.1 Design, operation and organization provisions

Borated water is stored in Refueling Water Storage Tank (RWST) and two Boric Acid Tanks (BAT).

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 on-site.

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 on-site.

5.2.2.3.2 Risks of cliff edge effects and deadlines

No cliff edge effects related to core recriticality after core damage had occurred were recognized, as described in the following chapter.

5.2.2.3.3 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 nuclear steam supply system 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.

There is no need to control either the injection flow rate or the boration of available water sources in order to ensure that the core can eventually be brought to controlled and stable state - although the core may not maintain its original geometry. Heat generated by a critical core can be removed if sufficient water can be provided to refill the reactor vessel.

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.

5.2.2.4 Prevention of basement melt-through: retention of the corium in the pressure vessel

5.2.2.4.1 Design, operation and organization provisions

See description in section 5.2.2.5.1.

5.2.2.4.2 Risks of cliff edge effects and deadlines

In the case that the vessel failure could not be prevented the best method to avoid CCI is to flood the containment basement and reactor cavity with water before RPV fails.

This is assured by wet cavity design and accident management measures by procedure SACGR-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 affects or dead-line were recognized concerning the basement melt-through due to the RPV failure.

5.2.2.4.3 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 5.2.3.1. The pump can provide limited core injection capability from RWST or BATs.

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 on-site.

If injection is not successful and relocation to the lower head occurs, reactor vessel 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 Core Concrete Interaction 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 CCI is to flood the containment basement and the reactor cavity with water, if possible before the RPV fails.

5.2.2.5 Prevention of basement melt through: retention of the corium in the reactor pit

5.2.2.5.1 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 on-site 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 on-site, 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 essential service water 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 on-site.

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 on-site. 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.

5.2.2.5.2 Risks of cliff edge effects and deadlines

Prevention of core-concrete interaction (CCI) 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 CCI 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 CCI 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 SACGR-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 affects or dead-line 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 CCI is endothermal 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.

5.2.2.5.3 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 Emergency Operating Procedures, as a response to the prolonged station blackout situation with significant damage to the on-site and off-site 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 CCI, 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.

5.2.3 Specific points

5.2.3.1 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 power plant and Krško, with the Brestanica's gas turbine "black start" ability (i.e. ability to start without external electric power). On complete loss of off-site power the engineered safety features 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 on-site, 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 on-site for diesel generators' 3-day operation.

It takes approximately 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 distributed around the nuclear island. Their 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 and power their respective Motor Control Centers (MCCs) by simply plugging the ready cable into the socket, which is mounted on the wall near the MCC. 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 on-site. 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 on-site, 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 on-site. 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 on-site, in quantity that allows a 3-day operation on full load for all the above described equipment.

5.2.3.2 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 on-site 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.

5.2.3.3 Availability and habitability of the control room

Habitability systems for the control room are designed so that habitability can be maintained under normal and accident conditions. To meet those goals, the following considerations have to be addressed:

- Maintain the required ambient air temperatures in all control room areas for the comfort and safety of personnel and to meet environmental requirements of the equipment.
- Adequately meet the requirements of 10CFR Part 50, Appendix A, General Design Criterion 19, to permit access and occupancy of the Main Control Room under the postulated accident conditions. There is adequate shielding provided for the Main Control Room to maintain desired habitability.
- To give the assurance of the possibility of the continuous presence in the vital plant arrears during and following the Design and Beyond Design Bases Accident.
- Provide a Control Room Emergency Charcoal Cleanup System for operation under abnormal conditions.
- Detect abnormal conditions such as high radiation, which requires isolation of the control room from the outside environment, in which case the Control Room Charcoal Cleanup System is operated.
- 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:
- Main Control Room Air Conditioning System,
- Main Control Room Charcoal Clean-Up System,
- Chilled Water Generating and Distributing System.

During post-accident conditions, pneumatically operated dampers automatically isolate the control building room from the outside atmosphere. The main control room air conditioning and electrical rooms cooling systems assume full recirculation. The main control room cleanup system is manually started on control room isolation signal, to keep the area habitable.

The control room isolation is initiated by either a safety injection signal or by a high radiation signal as detected by the radiation monitor in the main control room, or by isolation signal generated upon Hi-Hi Chlorine level detected on MCR-HVAC outside 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.

Although the possibility of fire in the control room is low, provisions were made to prevent recirculation of smoke-filled air to the control room. Smoke detectors for the main control room, relay and switchgear rooms, CRDM control room, and cable spreading areas would alarm the control room and the operator may fully open the outside air intake and the smoke relief dampers and fully close the return air damper, 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) form independent safety power bus. In case of loss of heat sink the system could work without the limitation (MCR Chiller System) due to the Air Cooled Chiller Water System. In case of loss of all AC power manual operation would be required to energize the minimum capacity of MCR-HVAC to keep the living conditions in the MCR adequate. Depending on the number of operable cabinets in the MCR, the temperature of the MCR can reach 49 °C, although that figure is not expected to bereached, therefore limiting maximum temperature.

In case air quality in the MCR worsens and operators cannot normally breathe, 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 on-site, together with diesel-powered charging compressor.

In the plant post-accident shielding review the action plan includes requirements to mitigate the consequences of accidents in which reactor core is severely damaged. It is related specifically to the item “Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems which may be used in post accident operations”.

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, however the MCR will be accessible without the limitation even in the case of the Sever Accident.

5.3 Spent fuel pit

Design provisions of spent fuel storage

<u>Spent fuel pit design and configuration</u>

Krško NPP Spent Fuel Pit (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 system (SFPCCS). The walls of the SFPCCS are made of concrete, 1.83 m thick and are additionally fitted with 6mm thick stainless steel liner plates to prevent leakage of water.

At Krško NPP site, spent fuel pit is 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 (Safe Shutdown Earthquake) and OBE (Operation Basis Earthquake) seismic loads meeting Seismic Category I requirements.

A spent fuel pit leak detection system is provided to monitor the integrity of the liners for the spent fuel pit, fuel transfer canal and cask loading area.

Criticality in the spent fuel storage area is prevented both by physical separation of fuel assemblies and the presence of borated water in the spent fuel storage pool and by the use of borated stainless steel absorber sheets in the new spent fuel storage racks.

Under normal operation conditions as well as for postulated accident conditions, including pool boiling and optimum moderation, old Region I and old Region II calculated neutron multiplication factors do not exceed 0.95, including mechanical and calculations uncertainties, with a 95% probability at a 95% confidence level. For old Region I as well as for old 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.

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. Spent Fuel Pool Boron Concentration of 2000 ppm is required by NEK Technical Specification. Geometry configuration of SFP is still criticality safe in the case of postulated non borated cooling water.

<u>Spent Fuel Cooling and Cleanup System</u>

The Spent Fuel Pit Cooling and Cleanup System (SFPCCS) is designed to remove the decay heat generated by spent fuel assemblies stored in the spent fuel pit 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. The SFPCCS is designed to remove the decay heat produced by 1694 spent fuel assemblies (phase 1 of SFP reracking) and 2321 (phase 2 of SFP reracking) in storage following the refueling. Heat is transferred from the SFPCCS through either heat exchanger to the Component Cooling System. Either heat exchanger SFAHSF02 or SFAHSF01 together with SFAHSF03 would be in operation.

If it is 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 spent fuel pit cooling system is capable of maintaining the spent fuel pit water temperature at or below 72.4 °C when the spent fuel pit heat exchanger SFAHSF02 is supplied with component cooling water at the design flow and temperature. Use of the heat exchanger SFAHSF01 together with SFAHSF03 would result in a higher temperature of the spent fuel cooling water, i.e., 73.5 °C.

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 spent fuel pit 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 spent fuel pit. To protect against the possibility, the spent fuel pit 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 spent fuel pit 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 spent fuel pit 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 spent fuel pit below the water level required for radiation shielding. A depth of approximately 3.05 m (10 feet) of water over the top of the stored spent fuel assemblies is required to limit direct radiation to 2.5 mR/hr (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 spent fuel pit, 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 on-site, 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 spent fuel pit reaches either the high or low level set points (15 cm above or 16 cm below the normal water level (115.00).

There is also alternate equipment (different power supply from battery or generator, local indication, wide range) available on-site, which enables supervision of SFP water level in the case of loss of spent fuel pool water without regard to its origin.

Walkdown

The operators perform walkdowns of the SFP area twice per shift to check the SF system, temperature and level.

Fuel handling building and spent fuel pit 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.

5.3.1 Measures for managing the consequences of a loss of cooling function for the pool water

Krško NPP is designed and operated on the concept of design basis event. In the safety area, licensee (i.e. Krško NPP) is responsible for providing assurance that is capable of withstanding design basis safety threat, and for taking reasonable measures to assure that available resources are used efficiently and effectively in response to beyond design threats.

Guidance for system normal operation is given in plant's specific System Operation Procedures (SOP). Each SOP consists of several sections. Section 5 describes how systems are operated for normal operation, and describes how the systems are prepared for different modes of operation. Subsections describe equipment configuration changes related to draining, filling, starting, etc. Section 6 describes action steps related to system abnormal operation in the cases related to system active components malfunction. System Operating Procedure for power supply provides guidance for emergency equipment power supply from emergency buses.

Guidance for action in case of losing cooling water pool is given in plant specific Abnormal Operation Procedures (AOPs) and Emergency Operational Procedures (EOPs). EOP's 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 Main Control Room alarm: Initial corrective actions and guidelines will be given to the operators by the appropriate Alarm Response Procedure which will further lead the operators to implement the recovery actions in accordance with the System Operation Procedure 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 Abnormal Operating Procedure Uncontrolled Loss of Rx cavity or Spent Fuel Pit.

The AOP recovery actions are structured in the following manner:

Heat removal from the spent fuel pit can be achieved basically in two ways: through spent fuel pit heat exchanger or through evaporation of water from spent fuel pit (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 spent fuel pit and there is no concern about criticality.

Alternative means for establishing spent fuel pit 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 spent fuel pit water surface.
- Providing water from fire protection hydrant network to the system for purification of spent fuel pit 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 carbonate mud pool with portable fire pump to the system for purification of spent fuel pit water surface.
- Pumping water from circulating water intake pool with submersible fire pump and fire track to the system for purification of spent fuel pit 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 spent fuel pit is decreasing even if makeup to the spent fuel pit 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 River Sava 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 spent fuel pit. This could be done to the skimmer connection through vale 13030 or directly with the use of fire protection hoses into the spent fuel pit. 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.

5.3.1.1 Before and after losing adequate shielding against radiation

Piping of the Spent Fuel Pit Cooling and Cleanup System (SFPCCS) is arranged so that failure of any pipeline cannot drain the spent fuel pit below the water level required for radiation shielding. A depth of approximately 3.05 m (10 feet) of water over the top of the stored spent fuel assemblies is required to limit direct radiation to 2.5 mR/hr (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 83,5 hours if 8.36 MW decay heat is considered (designed up to 8.7 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 Rx trip was initiated and in the case of the Station Blackout (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 25.

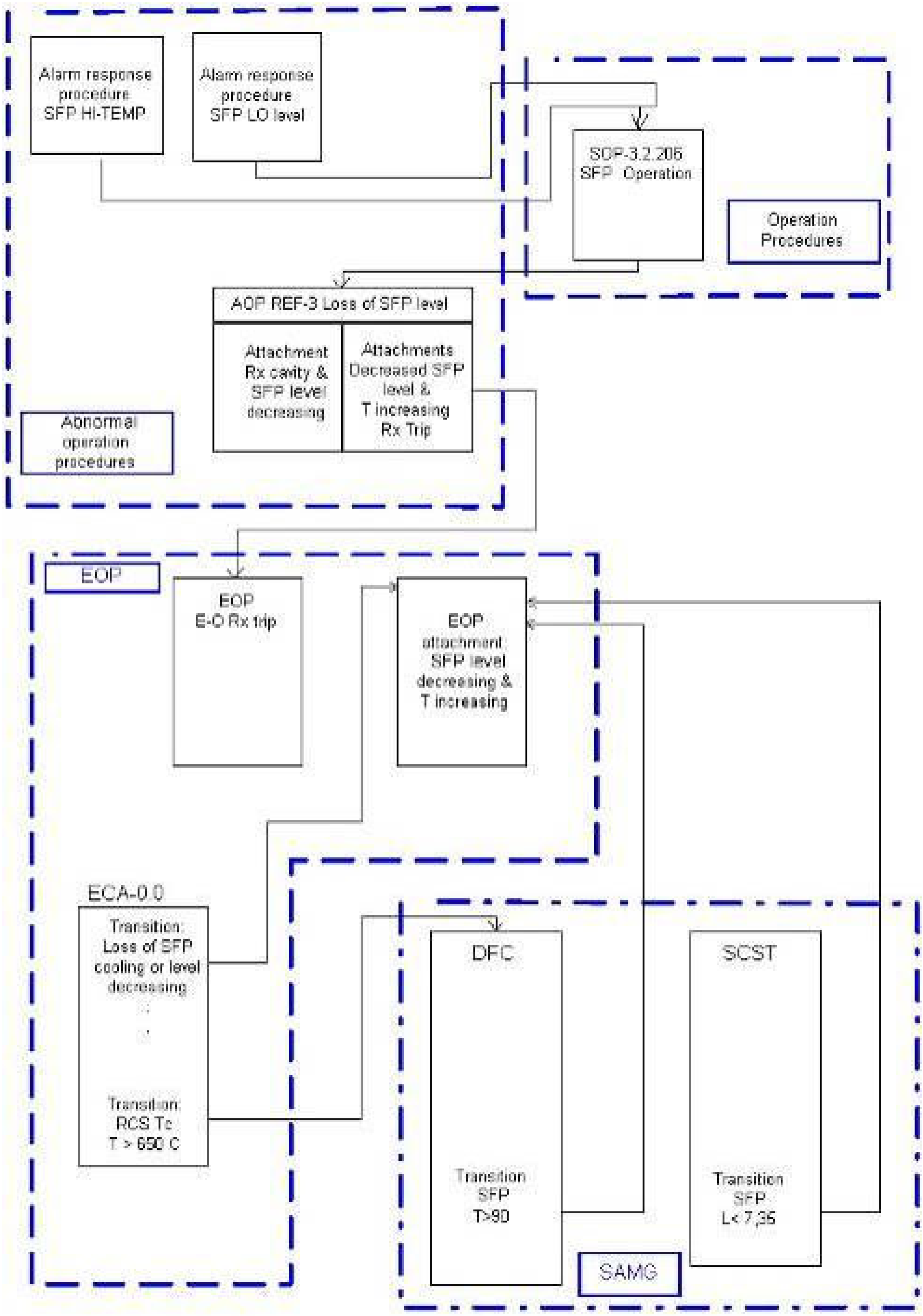

Figure 21: Procedural addressing of the SFP loss of cooling or level

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 technical support is leading the usage of this sets of procedures) will be guided to make transition to the specific SAG procedure which will address this issue to Loss of capability of SFP cooling is an Entry condition for the Severe Accident Guideline (SAG-5), 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 spent fuel pit is monitored in this Diagnostic Flow Chart and Severe Challenge Status Tree 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 Refueling Water Storage Tank 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 on-site. Details are described in section 5.3.1. The list of methods for establishing the SFP normal conditions is similar to the instructions in Abnormal Operating Procedures, however the procedure also cover the potential Station Blackout 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.

Spent fuel pit charcoal cleaning exhaust system function is to mitigate possible SFP radiological releases and removal of the released gases (also hydrogen).

5.3.1.2 Before and after uncovering of the top of fuel in the fuel pool

Mitigation and cooling restoration measures are the same as described in section 5.3.1.1.

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 approximately two days to reach the 3.05 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.

5.3.1.3 Before and after fuel degradation in the fuel storage facility

If spent fuel pit (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 spent fuel pit 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 Refueling Water Storage Tank 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 on-site.

5.3.1.4 Risk of cliff edge effects and deadlines

In the SFP hydrogen production is not expected or prevailed due to more likely oxidation of Zircaloy with atmosphere oxygen. If there is no uncover of the fuel elements degradation of the fuel cladding and radiological release is not expected also. 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 3.4 days to uncover fuel elements under estimated maximum decay heat (8.5 MW). Based o n 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.

5.3.1.5 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, AOP, EOP and Severe Accident Guidelines direct Operation/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 on-site. 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.

5.3.2 Specific points

5.3.2.1 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.

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 prolong up and above 16 hours.

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 Computer (PIS) system and with main control board as an alarm when the water level in the spent fuel pit 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 FA's 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 written to PIS system.

More challenging is to ensure indication during SBO and when level is dropping suddenly, constantly, due to evaporation leaking. 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 24V 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 Spent Fuel Pit boiling. The indication would raise when boiling starts.

Instrumentation that would not be useful until the Spent Fuel Pit cooling start is established is as follows:

- TI1409 – Outlet manifold of the spent fuel pit heat exchangers;
- TE1415 – Spent fuel pit heat exchanger SFAHSF01 outlet temperature;
- TE1416 – Spent fuel pit heat exchanger SFAHSF02 outlet temperature;
- TE1417 – Spent fuel pit heat exchanger SFAHSF03 outlet temperature.

Temperature Transmitters and Indicators TTI1415, TTI1416 & TTI1417 provide input signal for the computer Process Information System.

For online water temperature measurement, TE1421 is available.

TIS1400 – Spent fuel pit temperature. It measures temperature maintained in the spent fuel pit and gives a high alarm on the Main Control Board.

LS1401 – This instrument measures water level in the spent fuel pit. It generates high and low alarms on the Main Control Board. For online water level measurement, LE1420 is available.

LI1401 – Spent fuel pit measuring ruler measures water level in the spent fuel pit.

The internal video camera installed in the Fuel Handling Building connected to the plant computer network can be also useful for the supervising of the SFP area conditions.

5.3.2.2 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.

5.3.2.3 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.

6 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:

   i. Conclusion regarding the seismic core damage margin

      Based on the evaluation presented in chapter 2, seismic levels at which core damage would be likely are considered to be at the PGA range of 0.8 g to 0.9 g or higher.

   ii. 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 as high as 1.2 g or higher.

      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 to 0.9 g or higher.

   iii. 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 those seismic events with PGA in the range of 0.8 g to 0.9 g (or higher), at which reactor core damage is considered likely, were estimated to be very rare events at NEK site. Based on the plant specific studies, the return period for such an event is of the order of 100 000 years or larger.

2. Reactor core damage, and hence, a challenge to containment, can be avoided at water flows of the Sava river significantly higher than 7100 m3/s. Since the plant is shut down, the sequence development (to the point of core uncover) 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 m3/s flow, which is much higher than a 10000-year return period flow (4790 m3/s) or new revised PMF of 7081 m3/s. Having in mind that 1000-yr and 10000-yr 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 yr or larger.

3. In the case of Loss of the primary heat sink and/or loss of all AC, Krško NPP can:
    i. 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 on-site. In addition Krško NPP has also 5 portable engine driven generators on site to provide power to essential instrumentation.
    ii. Assure spent fuel pit cooling by adding cooling water with portable pumps. For the worse case (the entire core is unloaded to the spent fuel pit), the time available to establish water injection into the spent fuel pit is 53.7 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 83.5 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 on-site, 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 on-site.

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.
    i. On-site 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.
    ii. Off-site support and assistance to Krško NPP is provided by the local and other off-site 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 off-site support and up to 1 week with no needs for additional heavy mobile equipment from off-site. The mobile equipment essential for managing a severe accident (SAME) according to EOPs and SAMGs strategies is stored on-site. 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:
   i. Third independent diesel generator with a safety bus, which can be connected to both existing safety buses.
   ii. Provision to connect mobile diesel generator of capacity 2000 KVA to switch gear of the third diesel generator.
   iii. Flood protection upgrade (increasing the level of the dikes upstream of the site), 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:
   i. Acquiring (purchasing) on-site additional mobile diesel generator of capacity 2000 KVA.
   ii. Acquiring (purchasing) on-site additional pumping station to assure additional high capacity »portable water ring« around the plant - "HFS HydroSub 450 floating unit.
   iii. Installation of some additional quick connection points for mobile equipment.