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INTEGRATED APPROACH TO FIRE SAFETY AT THE KRŠKO NUCLEAR POWER PLANT — FIRE PROTECTION ACTION PLAN

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Abstract

Nuclear Power Plant Krško (NPP Krško) is a Westinghouse design, single-unit, 1882 Megawatt thermal (MWt), two-loop, pressurized water nuclear power plant. Construction of the plant was started in the mid 70's and initial criticality occurred in September 1981. NPP Krško is located on the north bank of the Sava River about 2 km southeast of Krško, Slovenia. The fire protection program at NPP Krško has been reviewed and reports issued recommending changes and modifications to the program, plant systems and structures. Three reports were issued, the NPP Krško Fire Hazard Analysis (Safe Shutdown Separation Analysis Report), the ICISA Analysis of Core Damage Frequency Due to Fire at the NPP Krško and IPEEE (Individual Plant External Event Examination) related to fire risk. The Fire Hazard Analysis Report utilizes a compliance — based deterministic approach to identification of fire area hazards. This report focuses on strict compliance from the perspective of US Nuclear Regulatory Commission (USNRC), standards, guidelines and acceptance criteria and does not consider variations to comply with the intent of the regulations. This review was constructed in accordance with the guidance set forth in Branch Technical Position CMEB 9.5-1, Appendix A. The probabilistic analysis method used in the ICISA and IPEEE report utilizes a risk based and intent based approach in determining critical at-risk fire areas. This method comprised of: Identification of potentially important fire areas; screening of fire areas based on probable fire-induced initiating events; each fire area remaining is numerically evaluated and culled on frequency; quantification of dominant areas. After the identification of these high risk areas, vital equipment affected by a fire in each area was assessed and modifications were suggested to reduce cdf (core damage frequency) for each area. Based on all above reports an extensive Fire Protection Action Plan was prepared utilizing the methodology for prioritization of proposed modifications as follows: CATEGORY 1 - CDF > 1.0E-6 event/rx-year and the potential modification(s) meets the cost benefit ratio criteria of <US$1000/person-rem to implement the modification(s). CATEGORY 2 - CDF > 1.0E-6 events/rx-year and the potential modification(s) exceeds the cost benefit ratio criteria of >US $1000/person-rem to implement the modification(s). CATEGORY 3 - CDF < 1.0E-6 events/rx-year.

NPP Krško has already completed the following suggestions/recommendations from the above and OSART reports in order to comply with Appendix R: Installation of smoke detectors in the Control Room; Installation of Emergency Lighting in some plant areas and of Remote Shutdown panels; Extension of Sound Power Communication System; Installation of a Fire Annunciator Panel at the On-site Fire Brigade Station; Installation of Smoke Detection System in the (a) Main Control Room Panels, (b) Essential Service Water Building, (c) Component Cooling Building pump area, chiller area and HVAC area, (d) Auxiliary Building Safety pump rooms, (e) Fuel Handling room, (f)

Intermediate Building AFW area and compressor room, and (g) Radwaste building; inclusion of Auxiliary operators in the Fire Brigade; training of Fire Brigade Members in Plant Operations (9 week course); Development of Fire Door Inspection and replacement program; sealing of Fire Barriers between areas; Development of Fire Response Procedures for improved response to fire events in critical areas of the plant. The above modifications, in particular the installation of smoke detectors in the Control Room, have substantially reduced the overcall plant fire-induced cdf. The most important remaining modifications to the plant include installation of a sprinkler system and fire wrapping of cables in some plant areas which will reduce the plant fire-induced cdf from 1.0E-4 events/rx-year to approximately 1.2E-5 events/rx-year which is equivalent to cdf values at US plants after implementation of Appendix R criteria.

1.0 Introduction

Nuclear Power Plant (NPP) Krško is a Westinghouse- designed, single-unit, 1882 megawatts thermal (MWt), two-loop, pressurized water reactor (PWR). Construction of the plant was started in the mid-1970s and initial criticality occurred in September 1981. NPP Krško is located on the north bank of the River Sava about two kilometers (km) southeast of Krško, Slovenia.

The purpose of the Fire Protection Action Plan (FPAP) is to prioritize proposed fire protection modifications contained in the NPP Krško Fire Hazards Analysis - Safe Shutdown Separation Analysis (SSSA) (Ref. 1), the International Commission for an Independent Safety Assessment (ICISA) Analysis of Core Damage Frequency Due to Fire at the Krško Nuclear Power Plant (Ref. 2), and the Operational Safety Review Team (OSART) (Ref. 3) reports using a risk-based approach which will provide for a timely reduction of the probabilistically-significant contributors to fire-induced core damage frequency (CDF). A cost benefit analysis has been performed for proposed modifications in fire areas, with the exception of the main control room, which were found to have fire-induced CDF exceeding 1.0E-6/ry (Ref. 4). The action plan utilizes event sequences and system models from the Krško IPE report (Ref. 5) and the resultant fire risk from the Fire IPEEE Level 1 and 2 reports (Refs. 4,6).

The SSSA report used a compliance-based, deterministic approach to identification of fire area hazards. The report focused on strict compliance from the perspective of United States Nuclear Regulatory Commission (USNRC) standards, guidelines, and acceptance criteria and did not consider variations to comply with the intent of the regulations. The review was conducted in accordance with the guidance set forth in Appendix A of "Guidelines for Fire Protection for Nuclear Power Plants" (Ref. 7). The purpose of the study was to perform and document a comprehensive analysis of the separation between redundant safe-shutdown components and cables in the context of post-fire shutdown system separation requirements defined by Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, 10CFR50, Appendix R (Ref. 8) and ancillary USNRC regulatory guidance.

The ICISA report used a probabilistic screening analysis method in determining critical at-risk fire areas. This method was comprised of the following steps:

(1) *Identification of potentially important fire areas:* Fire areas which have either safety-related equipment or cables were identified as requiring further analysis. Critical safety components required for hot standby and cold shutdown within these fire areas were identified. Areas not containing vital equipment were screened from further analysis.

(2) *Screening of fire areas based on probable fire-induced initiating events:* Estimation of the fire frequency for all critical plant locations and identification of the resulting fire-induced initiating events and "off normal" plant states were performed. Those areas with a random failure probability of less than 1.0E-3 were screened from further analysis.

(3) *Each fire area remaining was numerically evaluated and culled on frequency:* Fire area specific initiating event frequencies were used to screen the remaining areas with a low frequency of initiating events. The only fire areas remaining had contributions to CDF of greater than 1.0E-6/ry.

(4) *Quantification:* After the screening analysis eliminated all but the probabilistically significant fire areas, quantification of dominant areas was completed as follows:

> (a) The temperature response in each fire area was estimated for each postulated fire. The fire growth code COMPBRN (Ref. 9) with some modifications was used to calculate fire propagation and equipment damage.
>
> (b) A recovery analysis was performed which accounted for recovery of non-fire related random failures.
>
> (c) The probability of barrier failure for adjacent critical fire areas was evaluated.

The OSART review consisted of an international team of experts which performed a review of plant practices, including a review of the fire protection program. Detailed in their report were recommendations concerning improvements to the fire protection program associated with the
fire brigade, fire emergency procedures, fire barrier penetrations, training, fire cabinets, and fire brigade notification for response.
It has been found that implementation of Appendix R separation guidelines led to approximately one order of magnitude reduction in fire-induced CDF at United States light water reactors (Ref. 10). While Appendix R implementation has been found to substantially reduce fire risk on a plant-wide basis, most plant areas at NPP Krško (Ref. 4) have sufficient redundant methods of safe shutdown (i.e., critical safety-related equipment located in other plant areas). Therefore, Appendix R compliance is required in limited plant areas if the only goal is to accomplish a low fire risk.

2.0 Fire Protection Modifications Completed to Date

Many modifications have already been implemented at NPP Krško based on the recommendations contained in the SSSA, ICISA, and OSART reports. These modifications, in particular the installation of in-cabinet smoke detectors in the control room, have already led to a substantial reduction in risk (Ref. 4).

1. Installation of in-cabinet smoke detectors in the control room
2. Installation of emergency lighting at the evacuation panels and other plant areas

3. Inclusion of auxiliary operators in the fire brigade
4. Training of fire brigade members in plant operations (nine-week course)
5. Installation of a fire annunciation panel at the on-site fire brigade station
6. Development of a fire door inspection and test program
7. Installation of additional smoke detectors in many plant areas
8. Sealing of fire barriers between areas
9. Development of a modification package to upgrade and correct fire door deficiencies
10. Development of fire response procedures for those areas which were found to be the most risk-significant (Ref. 4)

3.0 Prioritization Method

The prioritization method for the remaining fire-related plant modifications is based upon providing for a timely reduction of the overall fire-induced CDF. This method ranks proposed modifications for a fire area into the following three categories:

1. Category 1 - CDF > 1.0E-6/ry and the proposed modification(s) meets the cost benefit ratio criteria of < U.S. $100,000/person-Sievert reduction (U.S. $1,000/person-rem reduction) to implement

2. Category 2 - CDF > 1.0E-6/ry and the proposed modification(s) exceeds the cost benefit ratio criteria of < U.S. $100,000/person-Sievert reduction (U.S. $1,000/person-rem reduction) to implement

3. Category 3 - CDF < 1.0E-6/ry

A CDF contribution of 1.0E-6/ry from a functional accident sequence (a combination of an initiating event together with functional failures resulting in core damage) meets USNRC reporting criteria for IPEEE analyses (Refs. 11,12). Functional accident sequences with frequencies less than 1.0E-6/ry are deemed to be insignificant contributors to risk. The cost-benefit ratio of U.S. $100,000/person-Sievert (U.S. $1,000/person-rem) is the USNRC regulatory value used in back-fit rule calculations and also employed for evaluating generic and unresolved safety issues.

For Category 1, the modifications are highly recommended to be implemented as soon as funding and resources are available. For Category 2, the modifications should be scheduled following completion of Category 1 modifications. For Category 3, the modifications are not significant contributors to fire-induced CDF and should be scheduled to comply with Appendix R criteria at plant management's discretion.

The cost analyses used are comprehensive and follow the guidelines of NUREG/CR-3568, "A Handbook for Value-Impact Assessment," (Ref. 13) and NUREG/CR-4627, Revision 1, "Generic Cost Estimates," (Ref. 14). The computer code FORECAST 3.0 (Ref. 15), which incorporates this knowledge-based information, was used to develop cost estimates for the proposed Category 2 plant modifications (with the exception of the main control room).

Cost analyses for the various tasks required for each proposed plant modification are performed according to standard engineering practices. This involves an initial design evaluation of the plant modification, identification of equipment and materials necessary for the modification, and an assessment of the work areas within the plant in which the proposed modification will take place. All plant cost estimates are presented in 1995 US dollars and represent implementation costs for the specific improvements, (i.e., one-time cost incurred by the nuclear power plant). There are no annual costs, (i.e., recurring costs), associated with any of the proposed modifications.

In addition to the cost of physical modifications, the cost analyses include costs for engineering and quality assurance, radiation exposure, health physics (HP) support, and radioactive waste disposal. Nuclear power plant costs associated with re-writing operating and testing procedures, staff training, and other technical tasks are also considered.

4.0 Core Damage Frequency (Level 1) Methodology and Results

The CDF quantification is based on the Individual Plant Examination of External Events (IPEEE) Fire Probabilistic Safety Assessment (PSA) which was completed in June 1996 (Ref. 4). The overall methodology used in the development of the Krško Fire IPEEE conforms with the guidance provided by USNRC Generic Letter (GL) 88-20, Supplement 4 (Ref. 11) and the detailed guidance provided in NUREG-1407 (Ref. 12). The fire PSA methodology followed the same approach used in the NUREG-1150 fire PSAs for the Surry and Peach Bottom plants (Ref. 16) and subsequent studies of the LaSalle and Grand Gulf plants (Refs. 17,18).

The methodology made use of past PSA experience (Refs. 19,20), generic databases, and other defensible simplifications to the maximum extent possible. The Krško Fire IPEEE is consistent with the Krško Individual Plant Examination (IPE) internal events analysis (Ref. 5) in that the same event trees, system success criteria, and recovery analysis assumptions were used.

The general methodology consisted of an initial plant visit, screening of fire areas to identify locations having the potential to produce risk-dominant fire sequences, and quantification. The initial plant visit is used to determine the general location of cables and components for the systems of interest, verify the physical arrangement of fire areas, and to complete fire area checklists which aid in the screening and quantification steps. The initial plant visit also includes confirmation from plant personnel that current documentation is being utilized, as well as clarification of questions which may have arisen during the walkdown. Finally, as part of the initial plant visit, a thorough review of fire-fighting procedures is conducted, including consideration of manual fire suppression by the fire brigade.

The screening analysis includes three sub-tasks. First, potentially important fire areas are identified. These are areas that have either safety-related equipment or power and control cables for that equipment. Fire areas are designated as portions of buildings that are separated from other areas by boundaries acting as rated fire barriers, except for certain outdoor areas which are provided with spatial separation from other fire areas. Fire barriers were defined based on the "as-built" condition of the plant. A large number of fire areas were screened out by inspection based on the absence of safety-related equipment or power or control cables for such equipment in the fire area.

The second step is to screen areas where fires would only lead to a fire-induced initiating event at a lower frequency than the corresponding internal events cause (Ref. 5). The third and final step in screening is to numerically evaluate the remaining fire areas and cull on frequency so that only fire areas which are potentially capable of yielding fire-initiated

TABLE 1. CORE DAMAGE FREQUENCY RESULTS, NPP KRŠKO (BEFORE AND AFTER FPAP IMPLEMENTATION)

Fire Area	Fire Scenario Description	Base Case CDF (Per Reactor-Year, Before FPAP)	CDF After FPAP (Per Reactor-Year)
CB-1	Fire-induced abandonment of the MCR (due to smoke obscuration), loss of feedwater, and either failure of recovery from the evacuation panels or a fire in the main benchboard from which recovery is not possible	8.7E-5	1.2E-5
AB-9	AB (El. 94.2 m) fire, fire-induced failure of component cooling system and the positive displacement charging pump, and random failure of either RCS cooldown or the turbine-driven AF system and the feedwater system	7.7E-6	<1.0E-7
CB-3A	Emergency switchgear room A fire, fire-induced loss of feedwater, fire-induced loss of both motor-driven AF pumps, random failure of the turbine-driven AF pump and feed- and-bleed cooling	4.1E-6	<1.0E-7
SW	Essential service water building fire resulting in fire damage to pumps A & B, and random failure of either the turbine-driven AF pump and the feedwater system, or RCS cooldown and the positive displacement charging pump	3.2E-6	<1.0E-7
AB-3	AB (El. 100.3 m) fire, fire-induced multiple spurious actuations of motor-operated valves powered from MCC 221 leading to an RCP seal LOCA and loss of instrument air; train A scenario (loss of instrument air) frequency is 2.1 E-7, train B scenario (RCP seal LOCA) frequency is 7.2 E-7	9.3E-7	9.3E-7
CC	Component cooling system building fire resulting in fire damage to pumps A & B, and random failure of either the turbine-driven AF pump and the feedwater system, or RCS cooldown and the positive displacement charging pump	8.4E-7	<1.0E-7

sequences with CDF contributions of greater than 1.0E-7/ry remain. A number of discrete initiating events were considered as appropriate, including loss of essential service water, loss of component cooling water, loss of instrument air, transients with and without main feed water, loss of a direct current (DC) bus, loss of offsite power, small loss of coolant accident (LOCA), and medium or interfacing system LOCAs due to spurious actuation of relief or isolation valves.

The quantification step involved detailed analyses of the potentially dominant fire-initiated accident sequences identified during the screening analysis. Quantification considered the temperature response in each fire area for each postulated fire, recovery analysis, and fire barrier failure analysis. Temperature response was modeled using the latest version of the COMPBRN fire growth code (Ref. 9). Recovery analysis considered, consistent with the internal events IPE, recovery of non-fire-related random failures. The barrier failure analysis considered potential combinations of adjacent fire areas which could result in core damage sequences.

Throughout the fire analysis, fire-related generic issues were addressed. Such issues were raised in the "Fire Risk Scoping Study" (Ref. 10) and in a report addressing Generic Issue 57 (GI-57), "Effects of Fire Protection System Actuation on Safety-Related Equipment" (Ref. 21). These issues include control systems interactions, total environment equipment survival, manual fire brigade effectiveness, inadvertent and advertent fire protection systems (FPSs) actuation, and seismic/fire interactions.

Six fire areas were found to have core damage frequency contributions of greater than 1.0E-7/ry before implementation of modifications. After implementation of FPAP modifications, only two such areas are identified. The pre- and post-implementation CDF results for NPP Krško are shown in Table 1.

5.0 Level 2 Methods and Results

The fire analysis containment event tree (CET) quantification methodology (Ref. 6) was consistent with the internal events IPE containment analysis (Ref. 5). The same bridge trees, containment system success criteria, CETs, and recovery analysis assumptions were used in the fire Level 2 analysis as in the internal events IPE.

The structure of the Level 1 fire IPEEE did not address containment heat removal (CHR) systems in the CDF quantification. Thus, it was necessary to construct a containment system tree (CST) or bridge tree to complete the system probabilistic analysis and to define the plant damage states (PDS) needed to perform source term analyses. The steps used to quantify the PDS are:

1. Quantify the CDF and obtain the dominant contributors
2. Construct the CST
3. Determine the success criteria for the top events in the CST
4. Define the PDS
5. Assign (and "bin" together) the dominant core damage sequences into the PDS
6. Link the containment systems fault trees to the dominant core melt sequences and quantify the frequency of each PDS

The PDS is a function of specific plant characteristics important to containment performance. These plant characteristics include the following:

1. Level 1 fire-induced initiating event (small loss of coolant, S, or transient, T)
2. Time of core melt (less than or greater than four hours; early, E, or late, L)
3. Core melt and reactor pressure vessel (RPV) failure pressure (high, H, or low, L)
4. Status of the emergency core cooling system (ECCS) (injection before reactor vessel failure, B; injection after reactor vessel failure, A; no injection, N)
5. Status of containment heat removal, addressing containment spray injection (CSI), containment spray recirculation (CSR), and RCFC (containment heat removal success, Y; containment heat removal failure, N)
6. Status of containment (initially intact, N; containment isolation failed, I)

The CET developed in Reference 22 was used for the fire scenarios. In addition, the source terms corresponding to each release category were maintained. Each PDS is processed through the CET. At each CET node a probability was developed which describes the confidence of the analysts that the event will or will not occur for the accident sequence under consideration. Once the nodal probabilities are developed for a PDS, the CET is quantified by multiplying the component probabilities and determining the probability of occurrence of

TABLE 2. NPP KRŠKO FIRE PSA LEVEL 2 RESULTS

Release Category Number	Release Category Definition	Base Case Release Frequency (/ry)	Release Frequency After FPAP (/ry)
—	Core Damage Frequency	1.0E-4	1.2E-5
1	Core Recovered In-Vessel; No Containment Failure	0	0
2	No Containment Failure	2.1E-6	3.2E-8
3A	Late (Beyond 24 Hours) Containment Failure, No Molten Core-Concrete Attack	3.5E-7	4.0E-8
3B	Late (Beyond 24 Hours) Containment Failure, Molten Core-Concrete Attack	8.1E-5	9.7E-6
4	Basemat Penetration (No Overpressure Failure)	6.8E-7	9.3E-8
5A	Intermediate (From End of Rapid Debris-Coolant Interaction Ex-Vessel to 24 Hours) Containment Failure, No Molten Core-Concrete Attack	8.5E-6	9.7E-7
5B	Intermediate From End of Rapid Debris-Coolant Interaction Ex-Vessel to 24 Hours)Containment Failure, Molten Core-Concrete Attack	1.2E-7	1.4E-8
6	Early (From Onset of Significant Zirconium Oxidation Reaction Until Vessel Failure and End of Containment Dynamic Response to Vessel Failure or End of Rapid Fuel Debris/Coolant Interaction Ex-Vessel) Containment Failure	1.7E-7	1.9E-8
7A	Isolation Failure, No Molten Core-Concrete Attack	1.2E-6	1.2E-7
7B	Isolation Failure, Molten Core-Concrete Attack	9.3E-6	1.2E-6
8A	Bypass, Scrubbed	0	0
8B	Bypass, Unscrubbed	0	0

TABLE 3. NE KRŠKO FIRE PROTECTION ACTION PLAN COST-BENEFIT ANALYSIS RESULTS

Fire Area	CDF (per reactor-year)	Suggested Modifications	Person-Rem Release Before FPAP	Estimated Cost	CDF After FPAP (per reactor-year)	Person-Rem Release After FPAP
CB-1	8.7E-5	Double fusing of control circuits for separation and add circuits to shutdown panels (minimal list)	42.0	by NEK	1.2E-5	6.0
AB-9	7.7E-6	Install area sprinkler system and one-hour wrap (nine critical cables)	3.8	$359,000	<1.0E-7	0.0
CB-3A	4.1E-6	Install three-hour cable wraps (nine cables)	2.9	$63,000	<1.0E-7	0.0
SW	3.7E-6	Install sprinkler system over SW motors, install one-hour heat shield between pumps and three-hour cable wraps (three power cables)	1.5	$101,000	<1.0E-7	0.0

each release category conditional on the occurrence of the given PDS. The process is repeated for all PDS.

The Krško fire PSA release categories are defined in Table 2. The table also shows the results of the PDS mapping by providing the frequency of the release category. Frequencies are shown for the "base case" (as-found condition of the plant) as well as for the case which will exist after implementation of the FPAP.

After implementation of the FPAP recommendations the release category frequencies are being dominated by the control room abandonment fire scenario. The only other remaining unscreened fire area AB-3 contributes less than eight percent to the total.

Four containment release categories were found to contribute 97.5% to the total CET frequency before modifications. After implementation of the modifications, three release categories contribute the same percentage to the reduced frequency total.

6.0 Level 3 Methods and Results

The MAAP4-DOSE code was used to calculate the consequences of the release categories, using site population and meteorological data. The resulting consequences, in person-rem per reactor-year, were multiplied by twenty to account for 20 years of remaining plant life. The risk results were calculated before and after implementation of modifications. The 20-year risk from fire-induced severe accidents before the modifications was 49 person-rem (0.49 person-Sieverts). After implementation of the modifications, the 20-year risk was 6.1 person-rem (0.061 person-Sieverts), a reduction by a factor of more than eight. The Level 3 and cost-benefit results are shown in Table 3.

7.0 Follow-Up Actions

Based on the recommendations in the Fire Protection Action Plan, engineering and design changes have been initiated and are in progress for the four Category 2 fire areas. The modification packages include design input/bases, safety evaluations, required FSAR changes, supporting analyses and calculations, design information and drawings, changes to plant procedures, and installation and test procedures. The general type of modifications involve circuit isolations, cable fire-wrapping or rerouting, and/or addition of sprinkler systems.

8.0 Conclusions

Using a risk-based methodology, the Fire Protection Action Plan evaluated and prioritized fire protection program improvements suggested in three independent reviews. The goal of the prioritization was to identify those improvements which provided the greatest risk reduction and assign the highest priority to such improvements.

Some plant fire protection program improvements and plant modifications have already been completed. These modifications, in particular the installation of smoke detectors in control room cabinets, have already substantially decreased the fire-induced CDF.

Based on the FPAP review and prioritization, the most important remaining modifications to the plant to reduce overall fire-induced risk are the Category 2 modifications for fire areas

CB-1, CB-3A, AB-9, and SW. These modifications are: (a) circuit isolation of vital equipment located on the evacuation panels and MCCs and the addition of RCS wide-range temperature and PORV control at the evacuation panels; (b) providing a three-hour fire wrap for train B cabling in fire area CB-3A; (c) installation of a sprinkler system and fire wrapping of cables in the AB basement; and (d) installation of a sprinkler system above and heat shield between the SW pumps, and fire wrapping of SW pump power cables. Implementation of these modifications will reduce the fire-induced CDF from approximately 1.0E-4 /ry to approximately 1.2E-5 /ry, which is equivalent to fire-induced CDF values at western nuclear power plants after implementation of Appendix R criteria. Design change packages to implement the Category 2 modifications are currently being developed.

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